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ANL/FPP/TM-125

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STARFIRE
A COMMERCIAL TOKAMAK REACTOR

An Interim Report

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Argonne, Illinois
McDonnell Douglas Astronautics Company
St. Louis, Missouri
General Atomic Company
San Diego, California
The Ralph M. Parsons Company
Pasadena, California

December 1979

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FOREWORD

This document represents an interim status report for the STARFIRE Project for the period of May to September, 1979. This period culminated in a two-day workshop held at ANL on October 31-November 1, 1979. The purpose of the workshop was to provide a mechanism for a broad representation of the fusion community to review the status of STARFIRE. A draft of this report was sent to workshop participants before the workshop.

A listing of workshop participants and a summary of their comments and suggestions are presented in the appendix. The STARFIRE team greatly appreciates the time and effort of the participants and, in particular, the chairmen of the three sessions: Dr. Dan Cohn (Plasma Engineering), Dr. J. L. Scott (Blanket Engineering) and Dr. Tom Shannon (Design Integration).

Since the workshop, the study has progressed with emphasis on many of the points raised at the workshop. The specific parameters and design features discussed in this report do not necessarily represent the final selections for the STARFIRE reference design. This process will be completed by mid FY 1980. The design is continuing to emphasize steady-state operation using rf current drive, the use of solid tritium breeder materials, and the application of limiter/vacuum pumping concepts for alpha particle removal. A final selection of blanket coolant and structural material will be made in the near future.

1.0 Introduction

1.1 Program Organization

The purpose of this document is to provide an interim status report on the STARFIRE project for the period of May to September, 1979. The basic objective of the STARFIRE project is to develop a design concept for a commercial tokamak fusion electric power plant based on the deuterium/tritium/lithium fuel cycle. The key technical objective is to develop the best embodiment of the tokamak as a power reactor consistent with credible engineering solutions to design problems. Another key goal of the project is to give careful attention to the safety and environmental features of a commercial fusion reactor.

This project was initiated in May, 1979, with the goal of completing the design study by October, 1980. The major emphasis of activities during FY 1979 has been to develop and compare options for key reactor design features. The plan is to select a set of reference parameters and major design features early in FY 1980 to provide a basis for an in-depth reactor and plant design during the rest of FY 1980.

The basic approach of this study has incorporated the following program elements:

- Multidisciplinary, multi-institutional design team.
- Fusion community input and review during the study.
- Utility and safety advisory committees.
- Interaction with ETF/INTOR groups.
- Incorporation of results and experience from previous tokamak reactor studies.

A major feature of the design approach for STARFIRE is the formation of a multidisciplinary design team made up of personnel from a national laboratory and industry. The program is being directed by Argonne National Laboratory and includes an industrial team led by McDonnell Douglas Astronautics Company, and including General Atomic Company and The Ralph M. Parsons Company. The project organization is shown in Fig. 1.1-1. The technical responsibilities of the respective organizations are shown in Table 1.1-1. In addition, the University of Wisconsin is providing support in the area of plasma impurity control.

Figure 1.1-1. STARFIRE Project Organization

An effort has been made during the early phase of the project to involve the fusion community in the review of major design decisions for STARFIRE. This has involved participation of outside groups in monthly project meetings and informal workshops on solid tritium breeders, helium-cooled blanket designs, and current driven steady-state operating modes. A major element in this process of fusion community input and review is the workshop to be held at Argonne National Laboratory on October 31 - November 1, 1979. Status reports on the STARFIRE project will also be presented at various technical society meetings during the next several months.

Advisory committees for STARFIRE have been established in two key areas. The first is a Utility Advisory Committee whose purpose is to provide input

Table 1.1-1. STARFIRE Task Responsibilities

	<u>ANL</u>	<u>MDC</u>	<u>GA</u>	<u>RMP</u>
Task 1. Reactor Design and Integration	S	P	S	-
First Wall/Blanket/Shield	P	S	-	-
TF Coil	S	-	P	-
EF Coils	P	-	S	-
Plasma Heating	P	-	S	-
Support Structure	-	P	S	-
Vacuum System	S	P	-	-
Cryogenic System	-	-	P	-
Thermal Storage	-	-	P	-
Tritium System	P	-	-	-
Impurity Control	P	S	S	-
Fueling	-	-	P	-
Instrumentation and Control	-	-	P	-
Electrical System	P	-	S	-
Plasma Engineering	P	-	S	-
R/D Assessment	P	S	S	S
Neutronics	P	-	-	-
Safety	P	S	S	S
Materials Selection	P	S	S	-
Maintenance	S	P	S	-
Systems Analysis	P	S	S	-
Task 2. Project Engineering	-	P	-	S
Subtask 2.1 Planning and Control	-	P	-	S
2.2 Plant Schedule	S	S	S	P
2.3 Licensing	-	S	-	P
2.4 Plant/Reactor Cost	S	P	S	S
Task 3. Balance of Plant	-	S	-	P
Subtask 3.1 Establish Requirements and utility interface	-	P	-	S
3.2 Plant Design	S	S	S	P
3.3 Cost Estimates	S	S	S	P
3.4 Plant Safety	P	S	S	S
Task 4. Information Plan	S	P	-	-
Subtask 4.1 Model	S	P	-	-
4.2 Film	S	P	-	-
- Data Book	S	P	S	S

P = Prime Responsibility

S = Support

from the utility perspective on commercial fusion power plants and to provide a mechanism for utility review of the design as it develops (see Section 1.2). Membership of the Utility Advisory Committee is shown in Table 1.1-2. Because of the importance of carefully considering the safety and environmental aspects of STARFIRE, a safety review committee has been established. Membership of the committee is shown in Table 2. This committee's primary purpose is to provide review and recommendations for the development of the reactor and plant conceptual design.

Table 1.1-2. STARFIRE Advisory Committees

Utility

Noel Amherd, Electric Power Research Institute
Fred Brunson, Union Electric Company
Ed Fiss, Duke Power Company
Robert Goodrich, Northeast Utilities
Betty Jensen, Public Service Electric and Gas
Bill Wolkenhauer, Washington State

Safety

J. L. Anderson, Los Alamos Scientific Laboratory
D. Cohn, Massachusetts Institute of Technology
J. G. Crocker, EG&G Idaho, Inc.
H. Howe, Jr., Princeton Plasma Physics Laboratory
M. Kazimi, Massachusetts Institute of Technology
L. Muhlestein, Hanford Engineering Development Laboratory
J. Powell, Brookhaven National Laboratory
T. Shannon, Oak Ridge National Laboratory

1.2 Utility Perspective of STARFIRE

The STARFIRE project has and will continue to actively solicit the inputs of electric utilities, the eventual customer for fusion power plants. The forum for inputs has been in the form of responses to an initial set of design and performance criteria and a questionnaire relating to broader questions of future power generation. A Utility Advisory Group, made up of representatives of six geographically distributed utilities has been formed to advise the project. Representatives from this group were convened in a two-day workshop in July, 1979. Their recommendations to the project are summarized in

the following sections. These recommendations are being incorporated into the study to the extent possible.

1.2.1 Utility Acceptance

The charter granted to an operating utility typically stipulates that the utility will produce reliable power on demand at the lowest possible cost. This places cost of electricity at the top of the list of utility evaluation criteria and imposes on fusion the requirement that it be less expensive than the viable alternative sources for generation. An additional requirement for acceptance of a new technology is that the utility be thoroughly familiar with the technology. This requires the involvement of utility and vendors during the development phase of the technology.

1.2.2 Design and Maintenance Considerations

Unit generating capacities of 3,000 to 4,000 MW thermal and approximately 1250 MW electric were recommended. The upper limit is dictated by the maximum capacity of a single turbine generator and as generators of greater capacity become available, the desired unit size may increase. The utilities suggested a 30-year economic life-of-plant and a 40-year design-life with consideration of reactor upgrade after 20 to 30 years. Steam is the preferred thermal conversion cycle. Cooling towers should be included to increase siting options.

A strong recommendation from the utilities was to attempt to reduce the field construction time. They suggested that fusion has a real opportunity to reduce the lead time and cost. The use of standardized designs, factory built and checked out is one example of an approach to reducing field construction time.

Remote, highly automated, maintenance should be planned for the reactor hall but provisions for contact maintenance should be included in the design. The contact maintenance would only be performed on a contingency basis since it is assumed that allowable dose rates will decrease in future years. Training simulators and reactor section mockups will be required for training and development of operations and maintenance procedures. The moving of large reactor segments (~ 300 tons) was not viewed as an obstacle to fusion acceptance. The scheduled maintenance plan should be similar to present

practice with an annual outage of 4 to 5 weeks and a 4-month outage every 5 to 10 years. Scheduled maintenance should be performed on a 24-hour basis during the scheduled outages.

1.2.3 Licensing

Tritium was of greatest concern in licensing and there was a strong recommendation for a primary coolant other than liquid lithium. The utilities recommended that the project develop licensing criteria and that a Preliminary Safety Analysis Report (PSAR) be developed for selected subsystems and plant design features.

1.3 Design Approach

The basic design guidelines for STARFIRE assume the successful operation of a tokamak engineering test facility (ETF) and a demonstration power plant. STARFIRE is considered to be the tenth plant in a series of commercial reactors. It is, therefore, assumed that a well established vendor industry exists and that utilities have gained experience with the operation of fusion plants.

The major areas of design emphasis in FY 1979 include developing plasma engineering characteristics of STARFIRE and selecting a reference blanket concept. The basic size of the reactor is determined by selecting the net power output and the first wall power loading. The net electric power output has been selected to be about 1150 MW(e). The first wall loading depends on the particular blanket concept (see Section 3.2) and is in the range of 2-4 MW/m². This results in reactor major radii in the range of 8 to 6 m. The toroidal magnetic field requirements are established by considering the plasma β and any toroidal field margin required for impurity and alpha particle control. The β limits have been taken to be generally consistent with those being adopted for ETF and INTOR. The plasma engineering considerations for STARFIRE are discussed in Section 3.1.

A major feature under consideration for STARFIRE is a steady-state operating mode based on a continuous plasma current drive. At this point, an rf lower hybrid current drive option has received the most attention (see Section 3.1). The potential advantages of steady-state reactor operation are numerous. In particular, such an operating mode is expected to greatly improve the overall reliability of the reactor. Based on

the current theory, it is believed that a steady-state operating mode is possible, but there is at present very little experimental verification.

Availability goals have been established as 85% for the reactor and 75% for the complete plant including the reactor. These goals provide a basis for design of maintenance equipment. The maintenance scenario incorporates the current utility practice of shutting down annually for one month and a four month shutdown approximately every 5-10 years.

An important design consideration is the choice of the plasma impurity and alpha particle removal concept. Several schemes have been considered including a limiter/vacuum pumping system, various types of divertors, gas puffing and plasma boundary flows, and additional margin in the toroidal magnetic field. Initial investigations indicate that modest pumping of helium with a limiter pumping system ($\sim 15\%$ of the alpha particle flux) coupled with about a 1.5 T margin in the maximum toroidal field may eliminate the need for a divertor, providing that a significant portion of the alpha particle energy can be radiated to the first wall rather than be deposited on the limiter. With respect to divertor options, particular attention is being given to developments for INTOR and ETF. In general, a nondivertor option is greatly preferred from an overall reactor engineering point of view.

Another key design consideration is the location of the equilibrium field (EF) coils. Consideration of major plasma issues are discussed in Section 3.1. The basic design approach is to locate almost all the EF coils outside of the toroidal field (TF) coils; all such EF coils would be superconducting. A limited number of segmented copper coils would be located inside the TF coils, but outside of the blanket and shield.

Safety has played a major role in considering various blanket options. In particular, efforts are being made to minimize the tritium inventory and the radioactivity induced in the first wall and structural materials in the blanket and shield. The design of the first wall and tritium breeding blanket is based heavily on a previous comprehensive review of possible blanket/shield designs⁽¹⁾ and is discussed in detail in Sec. 3.2.

Reference

1. D. L. Smith, et al., "Fusion Reactor Blanket/Shield Design Study," Argonne National Laboratory and McDonnell Douglas Astronautics Company, ANL/FPP-79-1 (to be published).

2.0 Overview of the Reference Design

A preliminary reference design has been identified for STARFIRE to serve as a focal point for the design trade studies. The design is modified as new requirements or trade study results are defined. The selection of the major parameters for the preliminary reference design point evolved from extensive system and trade-off studies as discussed in Sec. 2.1. A brief description of the key features of the preliminary reference design is given in Sec. 2.2.

2.1 Design Point Selection

Previous fusion reactor design and systems studies have demonstrated the presence of a wide range of design parameters and a diversity of design concepts. A primary goal of the STARFIRE study is to select, based on present knowledge, the most attractive set of design parameters and concepts that make tokamaks economically competitive and environmentally acceptable. In addition to experience gained from design and systems studies in the United States and worldwide, extensive tradeoff analyses were carried out to guide the selection process for STARFIRE. A primary tool for these tradeoff studies is a comprehensive systems computer program that is capable of predicting the performance characteristics and economics of the entire tokamak power plant. The ANL Systems Code⁽¹⁾ supplemented by the MDAC code⁽²⁾ was utilized for this purpose.

The major design parameters that characterize a tokamak reactor are the reactor power, the neutron wall load, aspect ratio, plasma elongation, major radius, plasma beta, magnetic field, scrape-off region thickness, and blanket/shield thickness. A brief review of the considerations that were factored into selection of these major parameters is presented below.

2.1.1 Reactor Power

It has been shown that tokamak reactors exhibit an economy of scale; i.e. larger power reactors have lower cost of energy. However, three considerations important to the utilities limit the desirable power rating of a plant. The first is the difficulty of raising the capital for larger power plants. The second relates to the cost of reserve electric power capacity that the utility must provide to compensate for scheduled and unscheduled outages.

The cost of reserve capacity increases with the size of the individual power plant. The third is the maximum capacity of a single turbine generator. Based on recommendations by the Utility Advisory Committee for STARFIRE, the most desirable power rating at present is in the range of 3000-4000 MW for thermal power and ~ 1250 MW for electrical power. Therefore, the fusion power for STARFIRE was selected as 3200 MW. This corresponds to a nominal thermal power of ~ 3800 MW, based on a 21 MeV per fusion reaction, and a net electric power of ~ 1150 MW. The recoverable thermal power will be modified by the addition of rf power for current drive and the loss of low temperature heat such as that in the limiter system.

2.1.2 Neutron Wall Load and Structure Life

A key parameter that has a substantial impact on the physical size of the reactor is the neutron wall load. The neutron wall load, P_{nw} , is related to the fusion power, P_f , as

$$P_f = P_{nw} A_w \left(\frac{17.6}{14.1} \right) = P_p V ,$$

where A_w is the surface area of the first wall, P_p is the average fusion power density in the plasma, and V is the plasma volume. For the same P_f , higher P_{nw} results in a smaller surface area, higher power density, smaller reactor volume and potentially lower cost. This underlines the motivation for developing designs with higher wall loads. Figure 2.1-1 shows the relationship between the major radius and the neutron wall load for $P_f = 3200$ MW and plasma elongation of 1.6.

There are limitations, however, on the ability to produce and use high wall loads. The upper limits on the use of high wall loads are dictated primarily by the first-wall cooling capability and the structure lifetime. Constraints such as the maximum operating temperature and thermal stresses place an upper-bound on the allowable wall load. For typical structural materials such as ferritic steels in pulsed reactor systems, the neutron wall load should be limited to ~ 2.5 MW/m² for helium coolant. Higher wall loads are possible with water and lithium coolants. In general, the maximum allowable wall load is higher for steady-state reactors.

For a given fluence lifetime, the neutron wall load should be limited so that the frequency of structure replacement is not excessive. In order to

Figure 2.1-1. Major radius as a function of neutron wall load at four values of the aspect ratio, A . Results based on: fusion power = 3200 MW, and plasma elongation = 1.6.

limit the fractional increase in the cost of energy due to the plant downtime for replacement of the structural material to δ , the structure lifetime must be sufficiently long to satisfy the following inequality:⁽³⁾

$$t_w > \frac{t_d}{365\delta} ,$$

where t_w is the structure lifetime in years and t_d is the total cumulative downtime in days for replacement of the structural material. For example, in order to limit the increase in the cost of energy to 10% (i.e. $\delta = 0.1$) when the downtime is 125 days the structure lifetime must be greater than 3.4 yr.

For a given structural material and a fluence lifetime, the loss of energy production resulting from choosing high P_{nw} and short t_w must be weighed against the economic gain realized by designing a small size reactor. Figure 2.1-2 shows the cost of energy as a function of the neutron wall load at two values of the integral neutron wall load, I_w , of 5 and 20 MW-yr/m² and at two different values for the total cumulative downtime, t_d , for replacement of the structural material. For $I_w = 5$ MW-yr/m²

Figure 2.1-2. Cost of energy as a function of neutron wall load. I_w is the integral neutron wall load in $\text{MW}\cdot\text{yr}/\text{m}^2$ and t_d is the total downtime in days for replacement of the structural material. Results are based on fusion power of 3200 MW, aspect ratio of 3.6, plasma elongation of 1.6 and $\beta_t = 0.067$.

and downtime of 125 days the neutron wall load should be kept in the range 2-2.5 MW/m^2 . For $I_w \sim 20 \text{ MW}\cdot\text{yr}/\text{m}^2$ the cost of energy decreases significantly as the neutron wall load is increased from 1 to 2 MW/m^2 . A smaller, but significant, saving in the cost of energy (COE) is realizable by increasing P_{nw} from 2 to 3 MW/m^2 . A slight change in COE is noticeable in the range $P_{nw} \sim 3\text{-}4 \text{ MW}/\text{m}^2$. The reasons for the modest increase in COE as P_{nw} is increased beyond $\sim 4 \text{ MW}/\text{m}^2$ will become evident from discussions later in this section.

It is clear from Fig. 2.1-2 that the achievable lifetime of the structural material has a significant impact on the cost of energy. By eliminating short plasma pulses and designing STARFIRE for steady-state operation, it is anticipated that a fluence lifetime of 20 $\text{MW}\cdot\text{yr}/\text{m}^2$ or greater is obtainable with selected candidate structural materials (see Sec. 3.2). There are several important advantages for such long life: (a) the cost of energy is substantially reduced because of less frequent replacement and higher

availability factor; (b) when the frequency of replacement is substantially reduced, the cost of energy becomes less sensitive to moderate variations in the downtime. This is quite important as it permits flexibility in the reactor design not available otherwise for designs driven primarily by the need for achieving very short downtime; (c) a less frequent replacement of the structural material results in a lower inventory of radioactive materials for which storage has to be provided. (It should be noted that even for the candidate structural materials with no long-term activation, adequate radioactive storage is necessary for two to five decades); and (d) the demand on material resources is less with longer life.

For a given fusion power, plasma elongation and aspect ratio, a higher wall load implies a higher plasma density, P_p . This varies as $P_p \sim \beta^2 B^4$. Since the plasma β is limited by stability considerations, a higher P_{nw} is obtainable only by providing a higher magnetic field as shown in Fig. 2.1-3. This figure shows the maximum toroidal magnetic field, B_m , required as a function of aspect ratio and neutron wall load. The two different scales for B_m on the left and right of Fig. 2.1-3 correspond to two different plasma impurity control schemes as discussed in Sec. 2.1.4. For $P_{nw} > 2 \text{ MW/m}^2$ there is $\sim 1 \text{ T}$ increase in the required B_m for every 1 MW/m^2 increase in P_{nw} . The economically attractive range for P_{nw} of 3-4 MW/m^2 requires a maximum

Figure 2.1-3. B_m as a function of A at several values of P_{nw} . The LHS scale is for $n_a/n_{DT} = 0.01$ and the RHS scale is for $n_a/n_{DT} = 0.15$.

toroidal field in the range 10-12 T which is considered acceptable for STARFIRE design as discussed shortly.

In light of the above considerations, a neutron wall load of 3.5 MW/m^2 has been selected for STARFIRE. This moderately high wall load appears to be a reasonable choice that makes it possible to design a relatively small size reactor without excessive requirements on first-wall cooling capability, maximum toroidal magnetic field, and frequency of structural material replacement.

2.1.3 Plasma Beta

Previous systems studies⁽⁴⁾ have indicated significant economic benefits for operating at high average plasma toroidal beta, β_t . However, the maximum realizable β_t is limited by plasma stability considerations. Based on recent theoretical analysis (see Sec. 3.1), the relationship $\beta_t = 0.24/A$, where A is the aspect ratio, has been assumed for the STARFIRE study.

2.1.4 Toroidal Magnetic Field

Figure 2.1-3 shows the maximum toroidal field (B_m) required as a function of aspect ratio and neutron wall load for $P_f = 3200 \text{ MW}$, $\kappa = 1.6$, $\beta_t = 0.24/A$, $\Delta_v = 0.1 \text{ m}$, and $\Delta_{BS}^i = 1.2 \text{ m}$. The scale on the left side of Fig. 2.1-3 shows the required B_m if an efficient plasma impurity control mechanism (e.g. divertor) is provided such that n_α/n_{DT} is ~ 0.01 . The scale on the right shows the B_m required if a less efficient impurity control mechanism (e.g. limiter/vacuum system, see Sec. 3.1) is utilized such that n_α/n_{DT} is ~ 0.15 . Figure 2.1-4 provides more details about the variation of B_m with the impurity level for the case of $P_{nw} = 3.5 \text{ MW/m}^2$, aspect ratio of 3.6 and $T_i \sim T_e = 14 \text{ keV}$. The rf power required to drive the plasma current decreases with T_e as discussed shortly. The dependence of B_m on T_e is shown in Fig. 2.1-5.

There exists at present considerable experience with NbTi superconductor. However, it has been shown that a magnetic field of $\sim 10 \text{ T}$ is the maximum practical limit for NbTi cooled to 4.2 K at atmospheric pressure. On the other hand, Nb_3Sn is capable of generating higher fields. Although present experience with Nb_3Sn is limited, the progress in the current technology development program indicates that the Nb_3Sn technology will be available in the STARFIRE time frame. Therefore, the 10 T limit of NbTi was not imposed as a constraint in the STARFIRE study; the required value of the

($P_f = 3200$ MW, $P_{nw} = 3.5$ MW/m²,
 $\kappa = 1.6$, $A = 3.6$, $\beta_t = 0.067$,
 $\bar{T}_i \sim \bar{T}_e = 14$ keV, $\Delta_{BS}^1 = 1.2$ m,
 $\Delta_v = 0.1$ m, $R = 7$ m)

Figure 2.1-4. Variation of the maximum magnetic field with the α -particle concentration, n_α/n_{DT} , and other impurity concentration, n_{Be}/n_{DT} .

($P_f = 3200$ MW, $\kappa = 1.6$, $A = 3.6$,
 $R = 7$ m, $\beta_t = 0.067$, $\Delta_{BS}^1 = 1.2$ m,
 $\Delta_v = 0.1$ m)

Figure 2.1-5. Variation of the maximum magnetic field and plasma current with the average electron temperature, T_e .

toroidal field was determined from engineering and economic tradeoffs for the overall reactor system.

2.1.5 Plasma Elongation

The elongation ($\kappa = b/a$) of a D-shaped plasma has a significant impact on the plasma performance, the reactor design characteristics, and economics. In particular, at higher κ the achievable β_t is higher but the required EF coil system becomes more complex and costly. Based on previous work, ^(1,5) a value of $\kappa = 1.6$ was selected for STARFIRE. This is believed to be nearly the upper limit on elongation if the important design goal of locating most of the EF coils external to the TF coils is to be achievable.

2.1.6 Power Requirements for Current Drive

Since STARFIRE will operate in a steady-state mode, a particularly important aspect of the design is the mechanism for plasma current drive. As discussed in Sec. 3.2, lower-hybrid rf is the selected option for current drive. Relativistic electron beams are also being examined as a backup option. An important impact of the current drive system on the reactor performance and economics is the electrical power requirements for this system. The dependence of the magnitude of this power for the rf system on key plasma and reactor design parameters is discussed below.

The theory of lower-hybrid wave-drive currents ^(6,7) indicates the ratio of rf power density to current density is proportional to the electron density. Thus, rf power is reduced by operating at higher plasma temperatures (lower densities for a fixed beta). However, as the plasma temperature increases above ~ 10 keV, the fusion power density starts to decrease and the ratio of rf power to fusion power is a minimum in the range 15-18 keV. ^(7,8) In surveying reactor operation at various temperatures, the desire to minimize rf power calls for considering temperatures above 10 keV, but larger toroidal fields needed to keep the total fusion power at 3200 MW must also be acknowledged at these higher temperatures (see Fig. 2.1-5).

In determining the dependence of reactor design on aspect ratio (A), for a fixed total power and wall load and for $\beta_t \propto A^{-1}$, the total current increases going to lower aspect ratio, while the major radius and plasma density decrease. Based on the analytic formula ⁽⁷⁾

$$P_{rf} = C R \bar{n}_e I ,$$

the rf power required to drive reactors for a series of equilibria in the range $3 \leq A \leq 4$ was computed. The coefficient C is a function of the plasma profiles, spectral width, and degree of current penetration; for typical reactor parameters, and for both centrally peaked and surface current density profiles, the rf power increases by 30-50% if the aspect ratio is reduced from 4.0 to 3.0. Hence, large values of A are preferred from the rf power point of view.

2.1.7 Major Radius and Aspect Ratio

With the fusion power and neutron wall load selected, the surface area of the plasma is defined. For a given κ , the aspect ratio (A) or the major radius (R) should be selected in order to fully define the plasma geometry. As shown previously in Fig. 2.1-1, at $P_{nw} = 3.5 \text{ MW/m}^2$, the major radius increases from 6 m to ~ 7.3 m if A is increased from 2.6 to 4.0.

The size of the reactor building, length of piping, etc. are strongly affected by the size of the reactor, in particular, by the value of $R + a$, where a is the plasma minor radius. Notice that the variation of $R + a$ as the aspect ratio is changed is less than the variation in R alone. Despite the reduction in the capital cost of several items sensitive to size when R is smaller, economic optimization does not necessarily favor the selection of minimum R . This is true for both pulsed and steady-state reactors although for different reasons.

For pulsed reactors, the optimum size is significantly affected by the central core radius, r_v . For a given magnetic field for the OH coil, decreasing r_v reduces the available volt-seconds and shortens the burn time resulting in a lower reactor electrical output.

For steady state reactors with no OH coil the problem of the central core radius disappears. In this case, the plasma current has to be driven by external means (e.g. rf or REB). If the electrical power requirement for the current driver were negligibly small, then the smallest aspect ratio should be chosen so that the major radius is minimum, provided of course that there is adequate space on the inner side of the torus to accommodate the TF coils and support cylinder. This is illustrated by the case $P_{rf} = 0$ in Fig. 2.1-6 which displays the cost of energy as a function of the aspect ratio.

Figure 2.1-6. Cost of energy as a function of aspect ratio for a steady-state reactor in two cases for the electrical power, P_{rf} , required for the current-drive system: $P_{rf} = 0$ and $P_{rf} = 502-84A$ where A is the aspect ratio.

As discussed earlier in this section, the electrical power requirement, P_{rf} , for the rf current drive system appears to be relatively large and increases with lower aspect ratio. Figure 2.1-6 shows the dependence of the cost of energy on aspect ratio for the case where $P_{rf} = 502 - 84A$, with P_{rf} in MW electric. In this case the cost of energy decreases as A increases up to $A \sim 4$. It is anticipated, however, that as a result of further work the rf power requirements will be reduced significantly. This would reduce the

optimum aspect ratio somewhat below $A = 4$. Therefore, $A = 3.6$ was adopted for STARFIRE. This results in a major radius $R = 7$ m. The maximum magnetic field required is nearly minimum at $A = 3.6$ as evident from results in Fig. 2.1-3.

A major incentive for considering relativistic electron beams (REB) as a backup option for plasma current drive is the much lower electrical power requirements compared to those needed for the rf system. Present estimates indicate that REB electrical power requirements are in the range 10-20 MW compared to ~ 150 -200 MW for the rf system. Assuming that the capital cost is roughly the same for both systems, the cost per unit power is $\sim 8\%$ lower with REB compared to that with the rf system. However, the present state-of-the-art for REB makes it difficult to quantitatively evaluate the technical and operational problems that might arise as the concept is investigated in more detail.

2.1.8 Inner Blanket/Shield Thickness

The thickness, Δ_{BS}^i , of the blanket and shield on the inner side of the torus, or more precisely the distance in midplane from the plasma side of the first wall to the location of the maximum toroidal magnetic field, has a substantial impact on the reactor size, the required strength of the magnetic field and reactor economics. A comprehensive investigation of the optimum value for Δ_{BS}^i has been carried out previously.⁽⁹⁾ The details of this previous work will not be repeated here, but the results are briefly stated.

For a given P_f , P_{nw} , β_t , and major radius, the advantages of a smaller Δ_{BS}^i are 1) lower maximum magnetic field and 2) larger central core (OH) radius, r_v . In pulsed tokamaks, the impact of the value of Δ_{BS}^i on r_v is large and becomes critical for low P_f , high P_{nw} designs. In a steady state tokamak with no OH solenoid, the primary incentive for reducing Δ_{BS}^i is to reduce the maximum magnetic field required.

On the other hand, there are several penalties for making Δ_{BS}^i too small. The increase in the radiation field when Δ_{BS}^i is decreased results in a) increase in the resistivity of the stabilizer material and a need for increasing the amount of the stabilizer to satisfy the cryogenic stability requirements, b) a decrease in the critical current density of the superconductor necessitating the use of more superconductor, and c) an increase in the heat generation rate in the TF coil since the increase in the power requirements for the

TF coil cryogenic system can be so large that the reactor net electrical power output is seriously reduced.

For the preliminary reference design of STARFIRE a value of $\Delta_{BS}^1 = 1.2$ m was selected based on careful considerations of the above trade-offs. This value includes $\sim 25\%$ void to account for the vacuum gap in the TF coil and the use of helium coolant in a portion of the inner blanket, if necessary. The shield consists of a combination of tungsten, boron carbide, lead and a structural material.

2.1.9 Outer Leg of the TF Coils

For a given major radius (R), plasma inner radius (a) and inner blanket/shield thickness (Δ_{BS}^1) the position of the inner leg of the TF coil is defined. In order to fully define the D-shape of the TF coil, the size of the vertical or horizontal bore must be selected. We will discuss this choice in terms of R_2 , the major radius of the midpoint of the outer leg of the TF coil. R_2 is the sum of the major radius, first wall minor radius, outer blanket/shield thickness (Δ_{BS}^0), clearance (Δ_c) in the midplane from the outer edge of the shield to the TF coil and half of the TF coil thickness. The choice of R_2 has a significant impact on many of the reactor characteristics and cost as discussed below.

With the reactor parameters previously defined in this section (R , a , Δ_v , B_m , Δ_{BS}^1) the only remaining parameters that affect R_2 are Δ_{BS}^0 and Δ_c . The necessary value of Δ_{BS}^0 varies with blanket and shield material and coolant choices. As a goal, the materials in the shield should be chosen to have inherently low long-lived radioactivity even if they are less efficient in radiation attenuation. For typical material choices, the required Δ_{BS}^0 is ~ 1.3 m and ~ 1.8 m for blanket with water (or lithium) and helium coolants, respectively. The clearance from the outer edge of the shield to the TF coils is required for several engineering reasons; the most dominant of which is to accommodate the coolant manifolds. The required R_2 is ~ 12 m for water or lithium coolants compared to ~ 13 m necessary for helium coolants.

The economics analysis shows a substantial penalty for increasing R_2 . The cost of energy increases by $\sim 3\%$ for each additional meter increase in the value of R_2 beyond 12 m. The cause of this penalty is that the value of R_2 directly influences the size and weight of the TF coils and their

support structure, the size of the reactor building, the length of the piping for the heat transport system, and the size of the externally located EF coils. For example, by increasing R_2 from 12 to 14 m, the ampere-turns and stored energy in the EF coils nearly double.

It should be noted that in order to keep the field ripple at the plasma to an acceptable level, the value of R_2 should be greater than a certain minimum. The minimum value for R_2 is larger for a smaller number of TF coils. A relatively small number, 12, of TF coils has been chosen for STARFIRE to enhance reactor maintainability. This makes extending the TF coils as a means of satisfying the field ripple criteria economically unattractive. Therefore, the alternative of an imposed field-ripple correction system (e.g. saddle coils⁽¹⁰⁾ or iron blocks⁽¹¹⁾) will be provided if necessary.

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2.2 Reference Design Description

An isometric drawing of the reactor design is shown in Fig. 2.2-1. The reactor cross section is shown in Fig. 2.2-2 and the design and operating parameters are given in Tables 2.2-1 through 2.2-8. The design has a major radius of 7.0 m and operates at a neutron wall load of 3.5 MW/m^2 . The plant delivers 1150 MW net electrical power. The reactor operates with a continuous plasma burn and develops $\sim 3800 \text{ MW}$ of thermal power. A 75% plant availability is planned over the 30 year plant life.

The reactor magnet system design includes twelve 11 Tesla TF coils. The EF coil system includes 4 segmented copper coils inside the TF coils for plasma stability control, but most of the EF coils are superconducting and located outside the TF coils for plasma equilibrium. The plasma startup, heating and current drive is accomplished by a lower hybrid rf system. This system eliminates the need for ohmic heating coils and minimizes the need for

Figure 2.2-1. STARFIRE Reference Design.

Table 2.2-1. STARFIRE Major Design Features

Net electrical power	1150 MW
Gross electrical power	1600 MW
Fusion power	3200 MW
Thermal power (nominal)	3800 MW
Thermodynamic efficiency (gross)	41%
Thermodynamic efficiency (net)	30%
Overall availability	75%
Average neutron wall load	3.5 MW/m ²
Major radius	7.0 m
Plasma half-width	1.94 m
Plasma elongation (b/a)	1.6
Maximum toroidal field (nominal)	11.0 T
No. of TF coils	12
Plasma burn mode	Continuous
Current drive method	rf
Plasma heating method	rf
TF coils material	Nb ₃ Sn/NbTi/Cu/SS
Wall structural material	Ferritic steel
Blanket structural material	Ferritic steel
Wall coolant	D ₂ O
Tritium breeding medium	Li ₂ O
Blanket coolant	Helium
Plasma impurity control	Low-Z coating + limiter and vacuum system + enhanced radiation + field margin
Primary vacuum boundary	At inner edge of shield

Table 2.2-2. STARFIRE Plasma Parameters

Major radius	$R = 7 \text{ m}$
Plasma half width	$a = 1.94 \text{ m}$
Elongation (b/a)	$\kappa = 1.6$
Plasma shape	"D"
Average toroidal beta	$\beta_t = 0.067$
Plasma current	$I = 11 \text{ MA}$
Safety factor (center)	$q_0 = 1.4$
Safety factor at edge	$q_a = 4.2$
Toroidal field on axis	$B_{to} = 5.62 \text{ T}$
Average DT ion density	$\overline{n_{DT}} = 1.0 \times 10^{20}$
Average electron density	$\overline{n_e} = 1.3 \times 10^{20}$
Average ion temperature	$\overline{T_i} = 17 \text{ keV}$
Average electron temperature	$\overline{T_e} = 22 \text{ keV}$
Fractional impurity concentration	$n_{Be}/n_{DT} = 0.04$
Fractional alpha concentration	$n_\alpha/n_{DT} = 0.09$
Fractional burnup	$f_b = 0.11$

Table 2.2-3. STARFIRE Impurity Control

Primary System

Low-Z coating

Limiter and vacuum system

TF margin to permit higher α and sputtered impurity concentrations

Enhanced radiation to reduce heat transport to limiter

Backup Option

Bundle divertor

(+ low-Z coating and enhanced radiation)

Major Parameters

Equilibrium impurity concentrations:

$$N_{\alpha}/N_{DT} = 0.09$$

$$N_{Be}/N_{DT} = 0.04$$

Effective reflection coefficient for He ≈ 0.70

Surface area of the limiter = $\leq 5\%$ of first wall area

Heat transport power to limiter = 115 MW

Maximum power density to limiter $\leq 11 \text{ MW/m}^2$

Limiter coolant = water

TF field margin $\approx 1.5 \text{ T}$

Enhanced radiation material = Kr or Xe

Radiated power to first wall $\approx 650 \text{ MW}$

Table 2.2-4. STARFIRE Plasma Current Drive and Heating

Heating method	rf lower hybrid
Current drive method (backup is REB)	rf
rf power to plasma	120 MW
Overall power efficiency	60%
Electrical power for rf	200 MW
Frequency, f	1.7
Index of refraction, n_1	1.3-1.6
Waveguide area	35.9
rf intensity	0.3

Table 2.2-5. STARFIRE Major Design Characteristics for First-Wall/Blanket

<u>First Wall</u>	
Coolant	D ₂ O
Structural material	Ferritic steel
Low-Z coating	Beryllium
Coolant exit temperature	300°C
Maximum structure temperature	400°C
Average neutron wall load	3.5 MW/m ²
Radiation power to the first wall	605 MW
Nuclear heat generation in first wall	250 MW
<u>Blanket</u>	
Breeding medium	Li ₂ O
Coolant	Helium
Structural material	Ferritic steel
Coolant exit temperature	500°C
Maximum structure temperature	525°C
Net tritium breeding ratio	1.05
Structure Lifetime Goal	8 yr
Blanket thickness	1.0 m

Table 2.2-6. STARFIRE Major Design Characteristics for the Shield System

Inner shield	W/B ₄ C/Pb/H ₂ O/ferritic
Outer shield	Ferritic/H ₂ O/B ₄ C/Pb
Inner blanket/shield thickness, Δ_{BS}^1	1.2 m
Outer blanket/shield thickness, Δ_{BS}^0	1.8 m
Maximum rate of atomic displacement in copper stabilizer (at $P_{nw} = 3.5 \text{ MW/m}^2$)	$6.69 \times 10^{-13} \text{ dpa/s}$
Radiation-induced resistivity in copper stabilizer (after 8 yr of irradiation at 0.85 availability)	$2.7 \times 10^{-6} \Omega\text{m}$
Maximum neutron dose to epoxy insulator in TF (after 30 years of irradiation at 0.85 availability)	$1.1 \times 10^9 \text{ rad}$
Maximum gamma-ray dose to epoxy insulator in TF (after 30 years of irradiation at 0.85 availability)	$2.7 \times 10^8 \text{ rad}$
Maximum nuclear heating rate in TF coils (at $P_{nw} = 3.5 \text{ MW/m}^2$)	$4.153 \times 10^{-5} \text{ MW/m}^3$

Table 2.2-7. TF Coils Selected Features

Conductor -	
Current:	24 kA
Superconductor:	Nb ₃ Sn (9-11 Tesla); NbTi (0-9 Tesla)
Stabilizer:	Copper
Configuration:	Cable
Field Grades:	4
Support Strip:	SS304, annealed
Insulation:	NEMA G-10
Cooling:	Bath, saturated He @ 4.2 K nominal
Type Winding:	Pancake
Helium Vessel:	Aluminum 6061T6
Vacuum Tank:	Aluminum 6061T6

Table 2.2-8. TF Coils STARFIRE Parameters

Peak Field:	11 Tesla
Number of Coils:	12
Dimensions:	
Height O.A.:	17.5 m
Outer Leg:	14 m mean radius at machine midplane
Centerpost Region:	
Height:	10.4 m
Vacuum Tank:	3.58 m O.R.
Coil:	3.43 m O.R.
Stored Energy:	67 GJ
Total Ampere-Turns:	189×10^6
Centerpost Region:	
Helium Vessel	
Outer Radius:	3.50 m
Inner Radius	~ 2.06 m
Radial Thickness:	1.44 m
Average Current Density:	750 A/cm^2
Total Heat Load (12 Coils)	
Nuclear:	10 kW
Disruption:	30 MJ

dielectric breaks in toroidal conductors. The shield incorporates dielectric breaks to prevent toroidal current flow and permit the EF coils that are located inside the TF coil to have adequate response time for plasma stability control.

The reactor shield is designed for life-of-plant and is not removed for normal maintenance operations. The shield concept is shown in Fig. 2.2-3. The reactor shield is water cooled and serves as the primary vacuum chamber.

Figure 2.2-3. Shield Design.

The shield protrudes between the TF coils at the top and bottom of the reactor to provide a vacuum duct to 24 cryosorption pumps. The vacuum duct opening through the shield is oversized to permit routing of coolant lines through the opening without significantly restricting pumping capacity. Each helium coolant line is shielded at the penetration through the shield by a bed of rods in a staggered pattern that limits neutron leakage. The shield contains poloidal dielectric breaks under every third TF coil. The dielectric joint, which is at the outside of the shield and completes the vacuum barrier, is exposed to about 10^{10} rads irradiation. The dielectric break will also be redundant and incorporate intermediate pumping. The shield pieces are designed to keep all seal welds between shield pieces in a single plane

without welded corners, in order to improve reliability and simplify the installation operation. The shield doors utilize redundant Buna-N seals to provide the vacuum barrier. The seals are located near the outside of the shield; however, degradation by exposure to tritium and neutrons is expected to result in the need for seal replacement each time the door is removed during maintenance.

The rf waveguides form a near continuous toroidal ring at the base of the reactor. Approximately 120 MW of rf power is required to start and drive the plasma. The power density at the plasma is 8 MW/m^2 . A mechanical rf waveguide connector is required to permit blanket removal.

Impurity control is achieved by toroidal limiters located in front of the vacuum slots. High-Z gas puffing is used to enhance thermal radiation from the plasma which reduces the limiter heat load to 7 MW/m^2 . As the plasma particles strike the limiter approximately 50% are directed to vacuum slots in the blanket where approximately 30% of those entering are pumped by the cryosorption pumps. The system extracts helium from the plasma at a rate which permits a continuous plasma burn. The limiter will be made of a high thermal conductivity material and will incorporate low temperature water cooling. The low grade heat in the limiter coolant will not be utilized in the power cycle. A candidate design utilizes water cooled, beryllium coated copper tubes. Other materials, which are more radiation resistant will be investigated.

The first wall and blanket are fabricated from ferritic steel. The first wall is cooled by heavy water (D_2O) to permit operation at a neutron wall load $P_{nw} = 3.5 \text{ MW/m}^2$. The blanket is cooled by helium because it provides high efficiency thermal conversion and because it does not produce an exothermic reaction with the Li_2O breeder in the event of a leak. The inboard blanket, which does not breed is constructed of ferritic steel and cooled by water. The first wall and inboard coolant water exits at 300°C and the blanket helium coolant exits at 500°C . The water is utilized for feedwater heating in a reheat steam cycle resulting in a thermal power conversion efficiency of 41%. The inboard wall is protected against plasma disruptions. The outboard wall is coated with beryllium to minimize the impact of sputtered impurities on the plasma. Tritium removal is provided by a separate low pressure, low mass flowrate helium loop. The overall net breeding ratio

of the plant is ~ 1.05 to achieve a doubling time goal of ~ 7 yr. Total tritium inventory in the plant is ~ 2.5 kg with ~ 1 gram in the low pressure helium purge stream.

The blanket system is divided into 24 wedge shaped sectors to obtain a size that permits removal of sectors between TF coils. Each blanket sector incorporates an rf wave guide, two limiter segments, and two stepped toroidal slots 20 cm wide for vacuum pumping. Each sector is mounted on an air bearing pad to permit removal and replacement.

The reactor building is metal lined to provide a boundary against tritium release to the atmosphere. Access doors are provided to the interior through airlocks to permit ingress of reactor components. Continuous cleanup of the reactor building air environment is provided. The building is designed for a maximum overpressure of 3 psi which could result from an accident in which the liquid helium from one TF coil and the coolant from one first wall/blanket sector are released into the building.

The power conversion system is planned as a standard steam cycle without an intermediate coolant loop. Preliminary evaluations of tritium permeation and potential overpressurization of the helium coolant system by a leaking steam generator tube have indicated that an intermediate loop is probably not required. High and low pressure turbines and wet cooling towers are used in the power conversion system (Sec. 3.3).

Other design options being investigated include: use of a bundle diverter for impurity control; use of relativistic electron and other plasma current drive techniques; and alternate first wall blanket coolants, breeder and structural materials.

3.0 Component Design Considerations

3.1 Plasma Engineering

3.1.1 Summary

The basic plasma parameters of the STARFIRE design are listed in Table 2.2-2. The reference operating point is a moderate β , moderate elongation D-shaped plasma, with the plasma current induced and maintained by a lower hybrid drive.

Impurity control is accomplished by several parallel techniques including a limiter/vacuum system which removes about 30% of the helium outflux from the plasma. The plasma density has been kept fairly low, in order to minimize the required rf power.

During the burn, the plasma will be controlled by the injection of moderate Z material, e.g. Kr, to radiate heat, and possibly by adjusting the rf power. The main equilibrium field is established by a system of superconducting EF coils located outside of the TF coils. Four internal coils are used to trim the plasma position and to control disruptions.

3.1.2 MHD Equilibrium

This section describes the considerations leading to the choice of the reference equilibrium operating point (Table 2.2-2). We have considered D-shaped plasma equilibria in toroidal geometry (as defined in Ref. 1 and 2) with pressure profiles characterized by a width parameter α . Using a diamagnetism function⁽²⁾ $F^2(\psi) = F_0^2 (1 - \delta \hat{\psi}^2)$ it is possible to generate a broad range of low and high β_t equilibria with various axis and limiter values of the safety factor, q_a and q_b , respectively. Defining the aspect ratio as A, elongation as κ , and triangularity as d, the highest stable β_t is in general a function of A, κ , d, α , q_a , and q_b . This functional dependence is under active investigation in the physics community, and the operating point for STARFIRE is based on a study of the literature presently available.

Stability to the $n = 1$ external kink and to ballooning modes increases linearly with κ up to $\kappa \sim 1.6$ and levels off for higher values.⁽¹⁾ Likewise, stability increases with d for $\kappa \sim 1.6$, up to $d \approx 0.5$.⁽¹⁾ Due to the difficulty of EF coil design for large d⁽³⁾ a moderate d value was chosen; the

reference values are $\kappa = 1.6$, $d = 0.2$. Broad pressure profiles (low α) appear to yield the highest β_t stable to ideal modes,⁽¹⁾ and for most of our considerations, we have chosen $\alpha \sim 1.4$. Steady-state auxiliary heating associated with the current-drive mechanism will yield some control over the pressure profile width, but α must remain somewhat uncertain until more detailed one-dimensional plasma transport calculations have been developed. The aspect ratio has been retained as an explicit variable in the system tradeoff studies performed, and, based on the trends of stability studies,^(1,4) the maximum stable β_t is taken as $\beta_t \propto A^{-1}$.

Since the plasma current is driven by auxiliary power supplies, it is desirable to minimize the total plasma current in order to reduce the required external power. Plasma stability appears to have a rather complicated dependence on the total current and current density profile [or, equivalently, the safety factor profile $q(\psi)$]. The variation in maximum β_t with q_a for a fixed shear [i.e., q_b/q_a and $q(\psi)$ held constant] is illustrated in Fig. 3.1-1 and Ref. 5 and 6. At large q_a stability is limited by ballooning modes with $\beta_t \propto q_a^{-2}$,^(5,6,7,8) whereas at low q_a (~ 1) stability is limited by both the $m = 1$ kink and internal modes.^(4,5,6,8,9) Near the turnover point, it is possible to increase q_a significantly (thus decreasing the plasma current) without much change in the maximum β_t , as indicated by Fig. 3.1-1. For the system studies with different aspect ratios, equilibria were computed with $q_a = 1.45$, $q_b = 4.35$, and $\beta_t = 0.2A^{-1}$, which appears consistent with Fig. 3.1-1 and the various other references.

Another question concerns the best choice of q_b/q_a . Investigations reveal increased stability to ballooning modes once q_b/q_a is reduced below a critical value;^(1,10,11,12) in addition, Ref. 9 indicates high β_t stability to internal modes with low q_b/q_a and low q_a . If STARFIRE is driven by lower-hybrid generated currents, the required auxiliary power is minimized by creating high current densities in low electron density regions,⁽²⁾ for example, near the plasma surface. Since high beta equilibria are accompanied by broad current density profiles peaked off axis,^(2,9,13,14) these low shear equilibria are a natural match for rf-driven currents. A concern at low shear is the instability of the $n = 1$ kink;^(1,9) this mode may be controllable with external feedback⁽¹⁾ or by simply tailoring the force-free surface current density.^(1,4,14)

Figure 3.1-1. Maximum stable beta vs. q_{axis} . [Based on G. Bateman and Y-K. M. Peng, Phys. Rev. Lett. **38**, 829 (1977)].

Finally, stability of resistive modes must be considered in light of the evidence which connects them to macroscopic plasma behavior in present-day experiments. Of major concern are the double-tearing-modes associated with hollow current densities and "double-valued" $q(\psi)$ profiles.⁽¹⁵⁾ The nonlinear evolution of this mode may flatten the local current density maxima and uniformly distribute current density throughout the plasma interior.^(16,17) On the other hand, hollow current densities may be generated with monotonic q profiles,⁽²⁾ in which case this particular instability does not exist. Other theoretical work⁽¹⁸⁾ suggests resistive tearing modes will be essentially linearly stable for the high temperatures typical of STARFIRE, which would signify the absence of plasma disruptions as well as hollow current density stability.

Continuing efforts are being made on the theory of collisionless tearing modes and various kinetic modes, so our confidence in choosing a "best" plasma equilibrium cannot be completely justified at present. However, based on the studies described, we have selected an equilibrium with $q_b/q_a \sim 2-3$, $q_a \sim 1.5-2$, $\beta_t \sim 0.24A^{-1}$ and a monotonic safety factor.

3.1.3 Current Drive Options

Lower hybrid waves constitute the reference driver due to the amount of previous effort invested into understanding this option.^(2,19,20,21) A prime alternative would be intense relativistic electron beams (REB). Calculations⁽²²⁾ indicate that small amounts of REB power (a few MW) are required to drive ~ 10 MA in a tokamak reactor. However, the beam trapping and plasma equilibrium theory have not been as well developed for this option as the rf option. More attention will be devoted to the REB concept in the future.

Other wave drive candidates include low frequency transit time magnetic pumping^(23,24) and high frequency ECR detrapping of electrons to enhance the bootstrap current.⁽²⁴⁾ The bootstrap current formed the basis for the steady-state reactor design (Mark I) studied by the Culham group,⁽²⁵⁾ and McNally⁽²⁶⁾ suggested diffusing α particles may provide the necessary seed current for the bootstrap effect. Several of these drivers may be attractive and deserve additional study.

Other steady-state tokamak designs have been proposed. Neutral beam driven currents⁽²⁷⁾ suffer mainly from low efficiency neutralizers for positive beams and from the large size and shielding difficulty associated with linear beam lines. The "continuous" tokamak described by Peng⁽²⁸⁾ shifts the engineering difficulties to the poloidal coil system, requiring periodic transfer of large plasma volumes into various chambers of the torus.

3.1.4 Lower Hybrid Heating/Current Drive

The ratio of fusion power to rf power ($\equiv Q$) required to maintain the steady-state toroidal current of a tokamak is limited by the plasma's accessibility to lower hybrid waves.^(2,21) In particular, low density plasmas are more easily penetrated and result in higher Q than high density plasma. This is the reason that, for a fixed β_t , relatively high temperatures (and low densities) are attractive for STARFIRE. However, for $\bar{T} \gtrsim 15$ keV the fusion power density decreases so quickly that designs become unattractively large. A related strategy to maximize Q ⁽²⁾ is to drive the highest current densities where the plasma density is low, i.e., near the plasma surface. Particular attention must be given in this case to the stability of such current density profiles.

Additional increases in Q result from going to narrow spectra waveguide arrays. From Ref. 2 we find $Q \propto n_L^{-2} \left[1 + \frac{\Delta n_{\parallel}}{n_{\parallel L}} \right]^{-1}$ where $n_{\parallel L}$ is the lower index of refraction of a rectangular spectrum of width $\Delta n_{\parallel}$. While the lower limit on $n_{\parallel L}$ is set by accessibility, the width $\Delta n_{\parallel}$ is determined by the waveguide (Brambilla) grill geometry. Thus we will try to maximize the number of waveguide elements per wavelength of grill in order to better define the wave and to narrow $\Delta n_{\parallel}$. The upper limit on this number is set by multipactor breakdown and cooling requirements, as described below.

A typical lower hybrid system will be outlined to demonstrate the design constraints. Assume the average magnetic field in the plasma is 3.9 T and that the waves must penetrate to a density of $2.1 \times 10^{20} \text{ m}^{-3}$ at which point the lower hybrid frequency in a DT plasma is 1.23 GHz. In order to avoid parametric instabilities, we choose to operate at about twice that frequency: $f = 2.4 \text{ GHz}$. If a central index of $n_{\parallel} = 1.4$ is desired, this corresponds to a wavelength (waveguide array periodicity) of 8.93 cm. We assume the minimum width waveguide wall thickness which can be cooled in a fusion environment⁽²⁹⁾ is 5 mm; by increasing the number of guides per 8.93 cm period, the narrow gap per individual guide decreases and the electric fields in the guides increase. Twelve elements (phased $\Delta\phi = 30^\circ$) appears to be the upper limit to avoid multipactor.⁽³⁰⁾ The resulting system calls for individual guide openings 0.33 cm x 10.92 cm with the short dimension aligned along the toroidal direction; three double banks of elements $\sim 2.7 \text{ m}$ long are mounted in each of the twelve torus sectors, flush with the first wall at the bottom of the plasma chamber. The wave launcher thus completely encircles the torus centerline in a rectilinear fashion with a total area $\sim 19 \text{ m}^2$ facing the plasma. If 120 MW of rf power is transmitted, the corresponding intensity, $\sim 0.8 \text{ kW/cm}^2$, may be low enough to avoid nonlinear plasma modification of the spectrum. Surface heating to increase T_e at the plasma edge may also be desirable to reduce nonlinear effects.

Of the two candidate rf sources, crossed-field amplifiers (CFA) and klystrons, the former may promise to operate at higher efficiencies (70-90%), but, due to the large amounts of power dissipated in the rf structure,⁽³⁰⁾ the CFA option may increase the engineering difficulties. The transmission line is envisioned to be high power pressurized waveguide with a power splitting labyrinth to provide the grill phase shifts. The system will be designed with the ECR layer in the high pressure region; the vacuum

window will be placed out of the direct line of neutron radiation.⁽²⁹⁾ Phase shifters will be included to provide spectral tuning for the reactor start up period.

The startup phase will initiate with plasma breakdown. Once a target plasma has been provided, the rf current will be ramped up by increasing the rf power; I_p scales nearly linearly with P_{rf} . The power required to maintain the steady-state plasma current has been shown to be sufficient to ignite the DT plasma,⁽²⁾ so a single rf system will serve to both heat and maintain the plasma.

3.1.5 Overall Power Balance

Since the ratio of fusion power to required rf power is maximized at high electron temperatures ($\bar{T}_e \sim 15\text{--}20$ keV) for the lower hybrid driven tokamak,^(2,21) it appears to be desirable to operate at temperatures slightly elevated relative to conventional tokamak designs even though the fusion power density, $\frac{\langle \sigma v \rangle}{T^2}$, is a maximum at lower average temperatures ($\sim 8\text{--}10$ keV). If the electron energy containment time is sufficiently short, the electron and ion temperatures may be decoupled ($T_e < T_i$); this "hot ion mode" of operation has been shown to benefit the lower hybrid driven reactor⁽²¹⁾ and will be considered in our future calculations.

The particle and power balance equations assume a peaked, rather narrow density profile and relatively broad temperature profiles ($\alpha_T < \alpha_n$),⁽²⁾ consistent with central fueling and significant surface heat input from the rf source. We will investigate ways to increase the fractional burn by selectively fueling the interior with tritium at a rate equal to the influx of D_2 gas puffed from the plasma surface.⁽³¹⁾

The impurity removal system is assumed to be somewhat inefficient with the result that a significant steady-state concentration of impurities and alphas must be contained. Since these impurities are included in the total plasma beta, the toroidal field must accordingly be higher than for a pure ($Z_{eff} = 1$) plasma, and this "field margin" has been included in our design calculations.

3.1.6 Impurity Control System

The function of the impurity control system is to limit the helium and sputtered impurity concentrations in the plasma so as to maintain an ignited equilibrium. At equilibrium the α 's must be removed at the same rate they are being produced; however, the plasma can still contain an appreciable, steady-state, α density. For a steady state reactor, the only means of removing α 's is to pump some or all of the plasma outflux. Unfortunately, any method of removing α 's will also remove tritium; too high a pumping removal efficiency will cause a large increase in the tritium tied up in the vacuum ports, pump and tritium handling system. There is, therefore, a tradeoff between α removal and tritium inventory.

Our basic approach is to use a system with fairly low removal efficiency, $\sim 30\%$. This is sufficient to maintain a stable operating point, while at the same time reducing the amount of pumped tritium.

The reference system chosen is a limiter/vacuum system which together serves to concentrate and pump some of the plasma particle outflux. In addition, the TF field coil will be designed with enough field margin to contain the excess pressure of the α 's and electrons. There are two other basic features of the impurity control system; low-Z coatings are used on all exposed surfaces; and the plasma is operated so as to maximize the heat radiated from it and to minimize the transported heat. This approach, therefore, results in four techniques working in unison. These are:

1. low-Z coatings,
2. magnetic field margin,
3. enhanced radiation, and
4. limiter/vacuum system.

Table 3.1-1 summarizes the key motivation, uncertainties and needed future work for these techniques. The intent of selecting the impurity control system is to have the simplest and most reliable system possible, and to minimize the tritium inventory of the reactor. A divertor is still under consideration as a backup option. The following is a brief description of the components of the limiter/vacuum system.

Table 3.1-1. Impurity Control Options for STARFIRE

<u>METHOD</u>	<u>WHY NEEDED (or desired)</u>	<u>KEY UNCERTAINTIES</u>	<u>FUTURE WORK</u>
a) Low Z Coatings	Self sputtering coefficient for bare wall material could be greater than unity. Reduce uncontrollable amount of line and recombination radiation.	Value of sputtering coefficients, in-situ recoating, erosion of coating on wall by change-exchange neutrals.	Select material
b) Magnetic Field Margin	Reduces requirements on the "active" impurity control system - permits less pumping and hence smaller T inventory for any system.	Technological and economic tradeoffs only.	
c) Enhanced Radiation	Radiation is easier to handle than transport power, for any system. May be needed in any event to achieve stable operating point.	Confinement times of plasma, radiation properties of high Z material, plasma profile stability.	0-D and 1-D code analysis
d) Limiter/Vacuum system	Simplest system, avoids the complexity of divertors.	Radial diffusion coefficients, heat load on leading edge, structural integrity.	Detailed mechanical design and analysis of pumping efficiency

Low-Z Coatings - The advantages of low-Z coatings have been discussed in many reactor studies, e.g., Ref. 32. Unless the plasma edge temperature can be kept very low (≤ 50 eV) and divertors with very high shielding efficiencies ($\geq 90\%$) can be realized, there may be too much radiation from the sputtered bare metal surfaces (Fe, V, etc.) to permit ignition.^(32,33) In addition, at normal edge temperatures, there is likely to be a runaway sputtering situation because the self-sputtering coefficients of the bare metal can be greater than unity.⁽³⁴⁾ Low-Z coatings avoid these problems; the self-sputtering coefficients are considerably less than one⁽³⁴⁾ and secondly, the low-Z ions will be fully stripped over all or most of the plasma, resulting in little or no line/recombination radiation.

A typical choice for the coating material is Be, but BeO, B₄C, C and others are also candidates. The coating is needed on both the limiter and the first wall, the latter due to charge exchange sputtering.

Among the uncertainties of low-Z coatings are the value of the sputtering coefficients, but this is not a major concern unless the values are substantially different from expected values. There will be a net erosion of the coating on the first wall, because only neutral hydrogen impinges on the wall (in steady state), but this is probably not serious unless the region of charge exchange activity is quite localized; otherwise a large surface area is involved. There should seemingly be little or no erosion of the limiter coating because, in steady state, as much beryllium impinges on the limiter as is sputtered. In fact, there could be a net deposition of beryllium on the limiter due to transfer from the wall coating to the limiter. This point needs further analysis. Based on a computed edge temperature of ~ 1500 eV, the sputtering coefficients of Ref. 34 and on the limiter model of Ref. 35, the Be concentration in steady state should be $\leq 4\%$ of the DT density.

Magnetic Field Margin - This concept refers to the idea of operating with a higher magnetic field than that needed for a pure DT plasma, and in so doing to permit a fairly high level of impurities. Both a higher toroidal field and a higher poloidal field/plasma current are required. The motivation of this scheme is to relax the requirements on the rest of the impurity control system, in particular to permit a lower impurity removal efficiency. The use of the field margin scheme is based on the

judgement that it may be easier to provide an additional 1-2 T of field than to achieve a high (> 50%) removal efficiency (in fact even higher field margins may be plausible, e.g. Ref. 36). As mentioned above, it may also be desirable to intentionally operate with a small removal efficiency, in order to increase the fractional burnup and minimize the T inventory in the reactor.

The value of field margin required to contain a given value of impurity content is given, for a β -limited plasma, and for equal ion and electron temperatures, by:

$$\left(\frac{B}{B_0}\right)^2 = 1 + \frac{3}{2} \left(\frac{N_\alpha}{N_{DT}}\right) + \frac{(Z+1)}{2} \left(\frac{N_Z}{N_{DT}}\right), \quad (1)$$

where B and B_0 are the required values of TF field with and without impurities respectively, N_α is the helium concentration, N_Z is the sputtered impurity concentration, and Z is the atomic number of the sputtered impurity.

The RHS of Eq. (1) is the ratio of the number of ions and electrons in the impure plasma to the number in the pure plasma. This is equivalent to the plasma pressure ratio. For the anticipated values of N_α and N_Z , the field margin $\Delta B = (B - B_0)$ is in the range of 1-2 Tesla, depending on the specific particle containment time, and the removal efficiency of the limiter/vacuum system. The increase in plasma current, for given values of the safety factor and β_p , is in the range of 1 MA. If the plasma were operated in the hot ion mode ($T_e < T_i$), then the field margin requirement would be even less.

The magnetic field margin concept involves little or no uncertainties in the plasma physics, but there are economic tradeoffs; namely the increased cost of the TF magnet, EF coil, and EF power supply. An analysis of these tradeoffs using the ANL parametric systems code showed only a modest increase in the cost of electricity for a margin of < 2 T. It should be noted, however, that this margin increases the required field from ~ 9 T to ~ 11 T. Thus, this impurity control scheme is the basic driver for selecting Nb_3Sn rather than $NbTi$ superconductors. As discussed in Sec. 2.1, this is acceptable for the time frame of STARFIRE.

Enhanced Radiation - This concept refers to enhancing the radiated power from the plasma and thereby minimizing the transported power. This is desirable because the radiation, which goes uniformly to the wall, should be easier to handle than transport power, which goes directly to the limiter (or divertor plate). Since there may not be a natural plasma power balance at the plasma temperatures of interest, enhanced radiation may not only be desirable, but it may also be an alternative to enhancing the diffusion by such means as ripple coils, or the like.

In an ignition plasma, the α heating and rf power must be balanced by radiation and transport:

$$P_{rf} + P_{\alpha} = P_R + P_{tr}.$$

For STARFIRE $P_{rf} + P_{\alpha} \approx 750$ MW. Utilizing the prescriptions used in the INTOR workshops for energy transport (empirical electrons, 1/3 neoclassical ions), the value of P_{tr} is approximately 150 MW. If no special steps are taken to increase it, P_R will be considerably less than the ~ 600 MW needed for ignition. As envisioned here, high Z material would be added to the plasma, in controlled amounts, to enhance the radiation to the desired level. The best material for this purpose is probably a noble gas because it will not condense on the exposed surfaces and because it is chemically inert.

A preliminary analysis has shown that the following approximate amounts of materials are needed to achieve thermal equilibrium.

<u>Element</u>	<u>Required Amount</u>
Argon Z = 18	2.5% of DT (~ 12.5 mg)
Krypton Z = 36	0.3% of DT (1.5 mg)
Xenon Z = 54	0.1% of DT (100 mg)

Based on this table, Argon is probably unacceptable because too much of it is needed (affecting the plasma beta), while Krypton and Xenon appear to be acceptable. Other concerns are the sputtering of the low-Z coating due to the noble gas and the radioactivity resulting from the activation of the gas by neutrons. These also seem acceptable for Kr and Xe, but will have to be examined more closely.

The radiation resulting from either of these elements should be reasonably uniform over the plasma, based on preliminary analysis using a coronal equilibrium model.⁽³⁷⁾ In addition, lower Z material could be used to further reduce the transport power by radiating selectively from the plasma edge; this could be a further help in reducing the heat load on the limiter.

A problem with this technique may be that of control because while the noble gas can be added quickly, it cannot be removed quickly. A solution may be to operate slightly sub-ignition using the rf power as the "control knob" to control the operating point.⁽³⁸⁾

Work planned in this area is a more detailed 0-D and 1-D code analysis of the radiation enhancement, profile stability and control problems associated with this technique.

Limiter/Vacuum System - This system has two parts; one or more limiters which receive all of the charged particle flux, and adjacent vacuum ports which receive some of the particle reflux from the limiter. The limiters essentially serve to concentrate the particle flux; if no limiters were used, the vacuum ports would have to occupy a huge percentage of the first wall to be effective.

Limiter Design - The limiter/vacuum port impurity control approach is shown in Fig. 3.1-2. It consists of two toroidal limiters each located in front of a 20 cm slot through the blanket. As alpha particles strike the limiter they are neutralized and approximately 60% of the resulting helium is deflected into the vacuum slot where 25% of those entering are pumped by the vacuum system. Considering the effect of multiple reflections between the port and the limiter, the net pumping probability is about 30%, according to preliminary calculations.

The limiter system must absorb 160 MW of power that is uniformly distributed over a 5.5 cm scrapeoff zone. This results in a leading edge heat load of $\sim 8 \text{ MW/m}^2$ and a heat load of 2 MW/m^2 over the slanted deflector surface. Because of the high heat load, the leading edge will be constructed of a high thermal conductivity tube, such as AMZIRC copper No. C15000. This alloy (OHFC copper plus zirconium) is suggested because it has higher creep strength than OHFC alone. Analysis of an OHFC tube for the GA TNS limiter⁽³⁹⁾ under similar conditions has shown that a 1.25 cm dia x 0.125 cm wall tube with 10 m/s cooling water resulted in a maximum metal

Figure 3.1-2. Limiter/vacuum duct concept.

temperature of 280°C. At the predicted stress and temperature, the OHFC copper would elongate approximately 1% in 2 years of operation. The deflector portion of the limiter is constructed of a double wall sandwich with water coolant. The smaller heat load on this portion of the limiter will result in significantly lower temperatures which should avoid the creep limitations of copper.

Further design and analysis efforts are required to quantify limiter performance, configuration, and lifetime. Both the time required to replace a limiter and its lifetime are critical to its viability in power reactor operations. A life goal of 4 years has been established for the limiter.

3.1.7 Equilibrium Field Coil System

The equilibrium field coil system for a tokamak reactor has several important functions in addition to its fundamental role of balancing the plasma hoop expansion forces. It also provides the fields that generate plasma elongation. Stability of the equilibrium to axisymmetric modes, interchange modes, and ballooning modes is also dependent on the equilibrium field coils through sensitivity to the shape factors of the equilibrium.

Vertical position control is another important function of the EF coil system. Plasmas with vertical elongation tend to be inherently unstable to vertical displacements. Similarly, changes in the plasma pressure or current density profile can result in radial drifts of the plasma position. These drifts or displacements, if uncorrected by currents in the EF coil system, can result in plasma disruptions through strong interactions of the plasma with the vacuum chamber wall.

In the STARFIRE design, it was decided that the main EF coils should be superconducting, and should all lie outside the toroidal field coil, in order to facilitate repair or replacement of the poloidal or toroidal coils. Some auxiliary coils will be used for control purposes. Although an outside coil system increases the required coil current in the assembly, maintainability and reliability of the EF coils should be far superior to an inside EF coil system. With this configuration, however, it is important to minimize the TF coil size. For example, equilibrium calculations with EF coils exterior to a toroidal field coil with the outer leg at 16.8 m (an earlier reference point) showed that a total of about 100 MA turns would be required in the EF

coils to support a 10 MA plasma current, major radius of 7 m, elongation of 1.6, and beta poloidal β_p of 1, if in fact the equilibrium could be generated at all. This large EF coil current implies very large local forces on the TF coil and very large stored energy, as well as great positional sensitivity of the plasma to the EF coil currents. When the coil outer leg is moved in to the current design value of ~ 13.5 m, the EF coil current is reduced to around 40 MA-turns, indicating the importance of minimizing the TF coil size.

In STARFIRE, the plasma current startup period will be of the order of 10 s. The power supply for the EF coils will then be sized to provide the EF coil stored energy in about that period. Rapid or sudden changes in the plasma position, due for example to modifications of the current profile by MHD activity, cannot be followed by the external EF coil system without oversizing of the power supplies and difficult constraints on coil manufacture due to the high voltage to be applied to the terminals. It is more practical to place demountable copper coils as near to the plasma as possible for the purpose of plasma control. These coils would be actively controlled and would carry little average current, the average current requirement being transferred to the exterior superconducting coils as rapidly as their power supply permits.

In the STARFIRE design, plasma breakdown and steady-state current drive are achieved through the application of toroidally traveling electromagnetic waves. Since no OH coil is used, closed conducting paths can be placed near the plasma. It is then reasonable to ask whether the plasma control can be done passively. The question of how near to the plasma they must be in order to provide stability can be answered by considering the equation of motion:

$$MR = \frac{\mu_0 I_p^2}{2} \left(\ln \frac{8R}{a} + \beta_p + \frac{\ell_1^{-3}}{2} \right) - 2\pi R I_p B_v,$$

where M is the total mass of the plasma, R is the major radius, I_p is the plasma current, a is the minor radius, ℓ_1 is the plasma internal inductance, and B_v is the total vertical field (applied plus induced). As the plasma moves, the poloidal flux between the plasma and the closed conductors is conserved, which means that an effective B_v is induced which opposes the plasma motion. If the motion is caused by a change from peaked to flat current

profile, then the change in inductance is $\Delta l_1 \sim 0.5$, and the plasma center displacement normalized by the minor radius is

$$\frac{\Delta R}{a} = \frac{R}{a} \Delta l_1 \left[\frac{4R}{a} \left(\frac{R}{d} - 1 \right) + 5.6 + l_1 \right]^{-1}$$

where d is the distance between the plasma surface and the conductor. For $\Delta R = 10$ cm, approximately the acceptable displacement for STARFIRE, d must be in the range of 1 to 1.5 m. From an engineering point of view, this may be difficult, so that the STARFIRE design calls for control conductors that lie beyond the blanket and shield and are actively controlled. A plasma diagnostic that senses plasma displacements and velocities will be required; magnetic pickup loops near the vacuum wall would be ideal, but radiation damage could be a severe problem.

The blanket and shield are segmented toroidally and insulated to facilitate the soak-in of the control fields. In the event of a plasma disruption, large voltages will be generated in the vicinity of the plasma. These voltages will break down the intersegment insulation of the blanket or the control coil insulation unless an alternative path for the current is provided. For STARFIRE a system of toroidal conductors with spark gaps will be designed for this purpose.

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3.2 First-Wall/Blanket

3.2.1 Introduction

The technological and design aspects of various first-wall/blanket concepts have been considered in the selection of potentially viable designs for STARFIRE. The objectives of the present study involve identification of key technological constraints of candidate tritium-breeding-blanket design concepts, establishment of a basis for assessment and comparison of the critical problem areas and design features of each concept, and development of optimized first-wall/blanket designs for STARFIRE. The major emphasis has been placed on the development of a blanket design that is safe and environmentally acceptable. The primary guidelines established to meet these criteria are low tritium inventory in the blanket, minimal long-lived activation products and minimal stored energy.

Since breeding of tritium is considered essential, and since lithium is the only viable tritium-breeding medium, lithium in some form is required in the blanket. On the basis of engineering and design considerations, liquid lithium provides many advantages for the tritium breeder; however, because of perceived safety problems associated with a liquid lithium system, the focus of the present study has been on the use of solid lithium compounds for breeding. The primary objective has been to assess the viability of a blanket design based on solid breeders. Technical evaluation of the safety problems associated with the liquid lithium system was not performed in the present study. Consideration was given to input from some members of the Safety Advisory Committee for STARFIRE, based on limited safety analysis work performed previously. Likewise, the use of helium coolant has been the major focus of the present study because of its perceived safety advantages over other candidate coolants, viz., water and lithium. A major consideration in the selection of structural materials is the residual activation and the potential for reprocessing the material within one human generation, i.e. ~ 20 -50 years. This problem area impacts both radioactive storage and materials resource requirements.

Section 3.2.2 summarizes the important materials properties and considerations for candidate first-wall/blanket structural materials, coolants,

tritium-breeding materials, and neutron multipliers. Based primarily on compatibility and operating temperature limitations, potentially viable materials options for first-wall blanket system are deduced. The neutronics considerations for these materials systems are presented in terms of tritium-breeding capability and residual radioactivity in Section 3.2.3. Generic safety aspects and tritium processing methods are discussed in Section 3.2.4 for candidate design concepts. Based on the above considerations, a reference blanket design has been identified. If irresolvable, technical or engineering problems are identified in the early phase of the project, the alternate and backup designs will be considered. The reference design, which is developed within the safety and environmental criteria presented, consists of a ferritic steel first-wall and blanket structure with a water-cooled first wall, a helium-cooled blanket, and Li_2O as the breeding material. The alternate blanket-design, which circumvents some of the design difficulties of the reference blanket design, is similar except that water coolant is used in the blanket as well as the first wall. The backup blanket-design concept, which takes advantage of the inherent simplicity of liquid lithium for both breeding and coolant, meets the low activation and minimal tritium inventory criteria but retains the perceived safety problems associated with liquid lithium.

3.2.2 First-Wall/Blanket Materials

Types of materials required for the first-wall/blanket region include structural material, tritium-breeding material, coolant, and possibly a neutron multiplier. The candidate materials considered for each application are summarized in Table 3.2-1 and the important property requirements are discussed in the following sections for each type of material.

3.2.2.1 Coolants

Three candidate coolants considered in the present study are helium, water (D_2O) and lithium. Pressurized water is considered for the most part, although steam and boiling water systems have also been assessed. The choice of heavy water (D_2O), as discussed in more detail in Section 3.2.4, is based primarily on deuterium permeation and tritium processing considerations. Table 3.2-2 summarizes important problem areas generic to the respective coolants, and indicates critical problem areas that may be cause for rejection.

Table 3.2-1. Candidate First-Wall/Blanket Materials

Coolants:	Helium, Water* (steam), and Lithium
Breeding Materials:	Lithium (liquid), Li-Pb (liquid), Li_7Pb_2 , LiAl , Li_2O , Li_2SiO_3 , and LiAlO_2
Structural** Materials:	Austenitic stainless steel (316), High nickel alloy (INCO-625), Titanium (Ti-6242), Vanadium (V-15Cr-5Ti), Niobium (FS-85), and Ferritic steel (HT-9)
Neutron Multipliers:	Beryllium, BeO, Lead, PbO, and Zirconium

* Water coolant considered is D_2O .

** Representative alloys used for property data base.

The major incentive for the use of helium is its inherent safety because of its chemical inertness, and the key problem areas involve leakage into the plasma chamber, relatively low first-wall load limits, and large manifold requirements. Reliable containment of high pressure helium in complex welded structures is difficult and allowable coolant leakage rates into the plasma are very small. Detection of minute leaks in a helium saturated environment may be even more difficult, particularly if the reactor must be shut down to leak check. Although some design flexibility exists, the allowable surface heat loads on a helium cooled first wall are considered marginal. Helium coolant requires relatively large ducts and manifolds, which affect blanket thickness, maintenance problems, and magnet size. The importance of these problems cannot be completely ascertained until a total reactor design is evolved.

Water is an attractive coolant because of its good thermal-hydraulic properties. The major problems associated with water coolant relate to its high pressure and reactivity with the candidate breeding materials. High

Table 3.2-2. Summary of Coolant Assessment for STARFIRE

Potential Rejection Criteria [*]	Major Design Problem	Major Economic Impact	Coolant and Property
X			<u>Helium</u>
		X	Leakage and leak detection in first wall
		X	High pumping power
			Low first-wall load limit
	X		Requires high coolant outlet temperature
	X	X	Large manifolds and ducts
	X		Relatively high pressure
	X		Difficult shielding
	X		Low thermal inertia (heat capacity)
	X		High coolant velocity
	X		Poor heat transfer coefficient
X X			<u>Water (D₂O)</u>
	X		High pressure
	X		Neutron moderator
	X		Low-temperature operation
			Reactivity with lithium
			Reactivity with Li ₇ Pb ₂
	X		Reactivity with breeding oxides
		X	Difficult tritium removal
X X			<u>Lithium</u>
	X		Solid at room temperature
	X		MHD effect on pumping
	X		MHD effect on pressure
			Reactive with air
			Reactive with water
	X		Difficult cleanup of leak
	X		Gas trapping in channels
		X	Difficult maintenance
		X	Intermediate coolant loop

* Rejection criteria may apply only to certain combinations of coolant/breeder/structure concepts.

pressure water separated from high temperature breeding materials by a large heat-transfer surface area poses a significant safety concern.

Lithium is an excellent heat transfer fluid and can also serve as the tritium-breeding medium, thereby simplifying some of the design and compatibility requirements. The major concerns regarding the use of lithium coolant relate to safety because of its reactivity with air and water and difficult cleanup in the event of a leak. The reaction with water evolves hydrogen, which is particularly hazardous.

3.2.2.2 Tritium Breeding Materials

Lithium in some form is the only viable option for breeding tritium. Liquid lithium and several solid-lithium compounds have been considered for the breeding material (Table 3.2-3). Molten salts have not been included because of their poor breeding capability. Table 3.2-3 summarizes some of the important properties of the candidate breeding materials. In most respects, liquid lithium is the best candidate. Its breeding characteristics are very good and the only products are tritium and helium. Tritium processing methods that will maintain the blanket tritium inventory at ~ 0.1 kg have been developed. Since liquid metals are typically not susceptible to radiation damage, several generic radiation damage problems are avoided. The high heat capacity and thermal conductivity are advantageous and the possibility of utilizing lithium for the coolant considerably simplifies many blanket design problems. A wide range of operating temperatures appear feasible. The major problems associated with the use of lithium relate to its chemical reactivity and compatibility and the resultant safety concerns. As stated earlier, the reaction of liquid lithium with water is highly exothermic and hydrogen is evolved. The reaction with dry air is also highly exothermic, however, hydrogen is not evolved. Liquid lithium/lead alloys, which have been proposed as possible candidate breeding materials, provide some safety advantage over liquid lithium; however, the thermophysical and thermal-hydraulic properties are much less favorable than those of lithium. The degree of reduction in reactivity, and hence improved safety features, does not appear to override the less favorable materials and engineering considerations. Therefore, liquid lithium is the only liquid metal option considered in the present study.

Table 3.2-3. Summary of Properties of Candidate Breeding Materials

	Lithium	Li ₇ Pb ₂	Li ₂ O	Li ₂ SiO ₃	LiAlO ₂
<u>Physical and Chemical Properties</u>					
Melting point, °K	459	999	1970	1470	1880
Density, g/cm ³	0.51	4.50	2.01	2.52	2.60
Li atom density, g/cm ³	0.51	0.49	0.93	0.36	0.27
Thermal cond. W/m-°K	~ 50	~ 20	~ 3	~ 3	~ 3
Stability	Significant vapor pressure above 600°C	Reduced solidus if not stoichiometric	Significant vapor pressure above 1200°C (sublimes)	Reduced solidus (~ 1025) if not stoichiometric	
<u>Chemical Compatibility</u>					
Helium coolant	Oxidizes	Oxidizes	impurity interactions		
Water coolant	Very reactive + H ₂	Very reactive + H ₂	Reacts + LiOH	Soluble - hot water	Soluble - hot water
Air (elevated temperature)	Very reactive	Very reactive	Reacts	Hygroscopic	Hygroscopic
<u>Neutronics</u>					
Breeding	Very good	Very good	Good	Requires neutron multiplier	Requires neutron multiplier
Activation	No long term	²⁰⁵ Pb	No long term	No long term	²⁶ Al
<u>Radiation Effects</u>					
Tritium breeding	Helium generation	Changes stoichiometry Lowers solidus temperature Helium production may cause swelling and T-retention	Forms LiOT	Changes stoichiometry Lower solidus temperature Phase instability	Changes stoichiometry Phase instability
Neutron damage	None	Swelling/densification (?) Reduced T-release Enhanced sintering	Swelling/densification (?) Reduced T-release Enhanced sintering	Swelling/densification (?) Reduced T-release Enhanced sintering	Swelling/densification (?) Reduced T-release Enhanced sintering

Three types of solid-lithium compounds have been considered, viz., the metallic compound Li_7Pb_2 , lithium oxide (Li_2O), and the complex oxides Li_2SiO_3 and LiAlO_2 . Although a few other compounds have been proposed in other studies, the compounds listed here are the favored compounds of each type. For example, LiAl is similar in many respects to Li_7Pb_2 ; however, the Li_7Pb_2 provides superior breeding capability because of the neutron multiplication of the lead and is the favored compound of this type. Major problems involving the use of Li_7Pb_2 as the breeding material relate to (1) its relatively low melting temperature, (2) its high reactivity with water and the evolution of hydrogen, (3) its oxidizing potential in air and with impurities in helium, (4) potential radiation damage problems, and (5) tritium release characteristics. As discussed in more detail in Section 3.2.4, the acceptable range of operating temperatures for adequate tritium release is only from ~ 320 to 390°C . This temperature range is not compatible with the requirements for helium coolant. Furthermore, direct contact of helium coolant with Li_7Pb_2 is not acceptable because of oxidation of the compound by impurities in the helium. Use of water coolant with Li_7Pb_2 in the blanket is not considered acceptable because of the safety problem. In most other respects, Li_7Pb_2 with water coolant provides an attractive breeder/coolant option.

Lithium oxide is the most attractive ceramic breeding material from neutronics considerations because of its relatively high lithium atom density. It is the only ceramic breeding material with which an acceptable tritium breeding ratio might be attainable without a neutron multiplier. In practice, the material must be used in a low-density form ($\sim 80\%$ theoretical density) with interconnected porosity to facilitate tritium release. As indicated in Section 3.2.4, the allowable temperature range of Li_2O for tritium release ($\sim 460\text{--}910^\circ$) is the largest of the candidate solid breeding materials. The effects of radiation on the solid breeding materials have not been well established. The predicted effects of radiation on sintering of the oxides are based on expected similarities with oxide fuel behavior in fission reactors. Stable oxides are typically well-behaved compounds and the internal heat generation in a UO_2 fuel pin is higher but similar to that expected in an Li_2O breeding blanket. An important consideration in the use of Li_2O relates to the probable formation of LiOT according to the breeding reaction:

Although tritium is released from Li_2O in uncondensable form (T_2) at low temperature, above $\sim 200^\circ\text{C}$ the tritium release is primarily ($\geq 95\%$) a condensable form, viz., LiOT or T_2O . The tritium release characteristics are probably considerably different for the two cases. Formation of significant amounts of LiOT , which melts at 450°C , may further enhance sintering. Release of highly corrosive LiOT to the coolant or tritium processing fluid may also present problems for the structural materials. Radiation-induced trapping of tritium will likely reduce tritium release rates at low oxide temperatures. Data for LiAlO_2 indicate release rates under similar conditions (low temperature irradiations) are reduced by nearly an order of magnitude for fluences of $\sim 10^{20}\text{cm}^{-2}$ compared to fluences of 10^{16}cm^{-2} .

In the present analysis it is concluded that Li_2O breeding material cannot be used in direct contact with high pressure helium coolant under anticipated conditions. Reaction of Li_2O with impurities such as H_2O in helium can lead to formation of LiOH and excessive mass transfer at high temperatures. This effect plus sputtering by high energy recoils and fragmentation of pellets caused by thermal-mechanical stresses could lead to flow blockage and overheating at various positions within a pellet-bed blanket. Since Li_2O has such a low thermal conductivity, it is particularly susceptible to high hot-spot temperature problems. Reaction of Li_2O with water at high temperature and high pressure will form LiOH , which again is liquid above 450°C and extremely corrosive. The reaction is exothermic under anticipated reactor conditions. High-temperature ($> 500^\circ\text{C}$) LiOH will corrode through most structural materials in a relatively short time (10's or 100's of minutes).

Two complex oxides, viz., Li_2SiO_3 and LiAlO_2 , are potentially viable breeding materials. In many respects their properties are similar to those of Li_2O . Because of their lower lithium atom density, a neutron multiplier is essential and enrichment of ^6Li is probably beneficial. The requirement of a neutron multiplier substantially complicates the materials compatibility problems in the blanket. The primary advantage of these two compounds over Li_2O is their lower reactivity with water. Of the two compounds LiAlO_2 is the more stable. According to the results given in Table 3.2-15, the predicted temperature range for acceptable tritium release is less for these compounds than

for Li_2O . The LiAlO_2 has a higher minimum temperature and Li_2SiO_3 has a lower maximum. The thermal conductivities and heat capacities of all of these oxides are similar, therefore, the thermal-hydraulic considerations are comparable.

3.2.2.3 Structural Materials

Six classes of alloys considered for the first-wall/blanket structure are austenitic stainless steel, high-nickel alloys, titanium alloys, vanadium alloys, niobium alloys, and ferritic steels. Molybdenum alloys were not included because welding and joining properties were considered unacceptable. Table 3.2-4 summarizes the major problem areas associated with each of the alloy systems. Included are the property requirements that are major factors in lifetime or operating temperature for each candidate alloy systems. The property requirements that may eliminate the alloy systems from further consideration are indicated. In certain cases, the potentially unacceptable factors apply only to specific combinations of structure/breeding material/coolant.

Austenitic stainless steels are the best characterized materials for fusion reactor service of any of the candidate alloys, and are a viable structural candidate with fairly well-defined and rather limited lifetime and operating temperature potential. On the basis of swelling, radiation embrittlement, and creep, the maximum operating temperature is limited to $\sim 500^\circ\text{C}$ with predicted lifetimes corresponding to $\sim 10\text{--}20 \text{ MW}\cdot\text{y}/\text{m}^2$ for limited loading conditions. Stainless steel is compatible with most of the chemical environments of interest; however, temperatures will be limited to $< 450^\circ\text{C}$ in lithium unless some method of corrosion protection is developed. Long-lived activation products and significant radioactive mass transport in water-cooled systems will be important design considerations. A major advantage is the ease of fabrication and welding which enhances the reliability of structural components.

The major advantage of high-nickel alloys over stainless steel relates to their better high temperature strength properties. However, primarily on the basis of radiation embrittlement due in part to the high-helium generation rate, the high-nickel alloys are not considered to be viable candidates for the first-wall/blanket structure. These alloys are unacceptable for use

Table 3.2-4. Summary Assessment of Candidate Structural Materials for STARFIRE

	SS	Ni	Ti	V	Nb	FS
<u>Physical/Mechanical Properties</u>						
Tensile	0	0	0	0	0	0
Creep-Rupture	X	0	X	0	0	X
Thermal Stress	X	X	0	0	0	0
Magnetic	0	0	0	0	0	X
TMT Sensitivity	X	X	X	0	0	*
Thermal Stability	0	X	0	0	0	X
<u>Radiation Effects</u>						
Swelling	*	X	?	0	X	0
Radiation Creep	X	0	?	?	?	X
Embrittlement	X	*	?	0	X	X
DBTT	0	0	?	0	0	*
<u>Compatibility</u>						
With Helium	0	0	X	*	*	X
With Water	0	0	0	*	*	0
With Lithium	X	*	X	0	0	X
With Hydrogen	0	0	*	0	0	0
With Solid Breeder	X	X	X	X	X	X
<u>Neutronics</u>						
Activation	*	*	0	0	*	X
Effect on Breeding	X	X	0	0	X	X
Helium Production	X	*	X	0	0	X
<u>Fabrication</u>						
Forming	0	0	X	X	X	0
Welding	0	0	X	X	X	X
Post-Weld Heat Treatment	0	X	0	0	0	*

* Serious problem area - potentially unacceptable.

X Design problem area.

0 Not a limiting factor.

? Insufficient data.

in liquid lithium systems and the long-lived activation level is even higher than for the austenitic stainless steels.

Titanium alloys have a large unirradiated data base and have been used extensively at temperatures to 500°C. However, the data base for irradiation effects on titanium alloys is insufficient to predict the operating limitations or probable lifetime. The maximum neutron radiation exposures are $\sim 4 \times 10^{21} \text{cm}^{-2}$ or about 2 dpa, which is two to three orders of magnitude below goal exposures. Also, compatibility with the hydrogen environment is considered a serious problem, particularly for the first wall. Major advantages of this system relate to neutronics considerations, e.g., low long-lived activation products and low-parasitic absorption. Reprocessing of titanium alloys after less than 20 years of storage may be feasible. Welding must be done in an inert environment and post-weld stress relieving is recommended for certain applications. Titanium alloys can be used with helium and water coolants. Allowable temperatures in helium and water will be limited by oxidation mechanisms. Titanium is compatible with lithium; however, the high solubilities in lithium of certain alloying elements, (e.g., Al and Sn) will probably limit the operating temperature of the alloys.

The vanadium alloys are not viable for systems with water or helium coolants because of excessive corrosion. Vanadium alloys will also oxidize in air, and therefore must be protected by an inert atmosphere. Welding must be conducted in a high-purity environment. Major advantages of vanadium alloys relate to their good radiation damage resistance, high-temperature strength, low long-lived activation products, favorable neutronics properties, and good compatibility with lithium.

In most respects the niobium alloy system is similar to the vanadium alloys. Niobium alloys are not compatible with water or helium coolants. Since they are less radiation-damage resistant than vanadium alloys and produce long-lived activation products, these alloys are generally less attractive than the vanadium alloys.

The major question regarding the feasibility of using ferritic steels for the structure relates to the effect of radiation on the ductile-brittle transition temperature. Initial results at relatively low fluences raise the ductile-brittle transition temperature above room temperature. Requirements

for post-weld heat treatment of all welds creates substantial fabrication difficulties. Attractive properties of the ferritic steels include relatively good radiation damage resistance, viz., low swelling and low-radiation creep. The induced radioactivity for the ferritic steels is less than austenitic stainless steel but considerably greater than vanadium and selected titanium alloys. Radioactive mass transfer may be significant in water-cooled systems.

3.2.2.4 Neutron Multiplier

Four materials considered for neutron multiplication in the blanket are listed in Table 3.2-5 with some of their important properties. Also listed are areas of concern regarding the use of each of the candidate neutron multipliers. Beryllium and BeO provide the highest neutron and energy multiplication, but beryllium is resource limited. Lead oxide is presently considered as the prime candidate. This compound circumvents the solid/liquid phase transformation at 327°C with lead, which poses problems because this temperature is in the typical operating temperature range. Major design problems with lead oxide relate to its low thermal conductivity, high thermal expansion coefficient, phase transformation at 540°C, compatibility with structural materials, and relatively high density.

Table 3.2-5. Candidate Neutron Multipliers

Material	Important Properties	Major Design Problems
Be	m.p. 1277°C; High thermal conductivity High n,α cross section	Limited resource Toxic material
BeO	High temperature ceramic Good thermal conductivity High n,α cross section	Limited resource Toxic material
Pb	m.p. 327°C High density	Phase change (liquid/solid) Compatibility with structure
PbO	m.p. 888°C Low thermal conductivity High thermal expansion coefficient	Phase transformation 540°C (~ 20% ΔV) Compatibility with structure Radiation swelling/den- sification

3.2.2.5 First-Wall/Blanket Materials Options

Based on properties of the candidate materials discussed previously, several combinations of coolant/breeder/structural materials result in potentially viable blanket design options. Table 3.2-6 summarizes the combinations of materials that possess some potential for a credible design option. The primary operating limitations are indicated for the various materials combinations, and where appropriate, candidate neutron multipliers are given.

Lithium is the only breeding material considered with lithium coolant since there is no reason to use another breeding material if lithium is present in the system. Helium can be used with all of the breeding materials considered; however, Li_7Pb_2 is the least likely candidate because of the conflicting temperature requirements of this coolant/breeder system. Water coolant is not acceptable with lithium or Li_7Pb_2 because of safety considerations related to the high reactivity and the emission of hydrogen.

From the list of materials options in Table 3.2-6, the most viable combinations of coolant/breeder/structure/neutron multiplier are given in Table 3.2-7. A design flexibility of different coolant and structure is considered acceptable for the first-wall and blanket if desirable. The reference blanket system is selected on the basis of perceived safety advantages associated with helium coolant and the solid breeding materials. Helium is selected only for the blanket coolant since water provides several advantages as the first-wall coolant. The ceramic breeding materials with their high temperature properties are most appropriate for the helium coolant. Lithium oxide is proposed as the breeding material since it is the only ceramic with potential for breeding without a neutron multiplier. The ferritic steels are compatible with both helium and water coolant and are selected for both first-wall and blanket structure. A low pressure helium purge stream over the Li_2O is used for tritium processing since direct contact of high pressure helium coolant and Li_2O is not acceptable. Table 3.2-8 summarizes the favorable and unfavorable features of the reference blanket system.

Since the thermal-hydraulic characteristics of pressurized water are superior to those of helium in temperature-limited systems, an alternate first-wall/blanket materials option is proposed. The alternate concept utilizes D_2O coolant in both the first wall and the blanket. Lithium oxide

Table 3.2-6. First-Wall/Blanket Materials Options Considered for STARFIRE

Coolant	Breeder	Structure *	Neutron Multiplier	Operating Limitations *
Lithium	Lithium	V-alloy	NR **	Mechanical properties of structure
		FS, SS, Nb-alloy	NR	Mechanical properties of structure, radiation effects on structure.
		Ti-alloy	NR	Coolant/structure compatibility, mechanical properties structure
	Li ₇ Pb ₂ , Li ₂ O Li ₂ SiO ₃ , LiAlO ₂			No reason to use other breeding materials with lithium coolant
Helium	Lithium	FS	NR	Mechanical properties of structure
		SS	NR	Radiation effects on structure; breeder/structure compatibility
		Ti-alloy	NR	Structure/breeder compatibility, mechanical properties of structure
	Li ₇ Pb ₂	FS, SS, Ti	NR	Tritium release from breeder; breeder/structure compatibility
	Li ₂ O	FS	NR	Mechanical properties of structure; tritium release from breeder
		SS	NR	Radiation effects in structure; tritium release from breeder
		Ti-Alloy	NR	Mechanical properties of structure; tritium release from breeder
	Li ₂ SiO ₃ , LiAlO ₂	Same as for Li ₂ O	BeO, PbO	Same as for Li ₂ O breeding material plus: compatibility of PbO with structure; melting of PbO.

* In all cases it is assumed that coolant, breeder, and neutron multiplier are not in direct contact with each other.

** NR - not required.

Table 3.2-6. (Continued)

Coolant	Breeder	Structure *	Neutron Multiplier	Operating Limitations *
Water	Li, Li ₇ Pb ₂			Coolant/breeder not compatible from safety considerations
	Li ₂ O	FS, SS, Ti-alloy	NR	Tritium release from solid breeder; corrosive effects of coolant/breeder interactions
	Li ₂ SiO ₃ , LiAlO ₂	FS, SS Ti-alloy	BeO, PbO	Tritium release from solid breeder; compatibility of PbO with structure;

Table 3.2-7. Selected First-Wall/Blanket Materials Options

Coolant	Coolant			Structure *		Neutron Multiplier
	FW	Blanket	Breeder	FW	Blanket	
Reference	D ₂ O	He	Li ₂ O	FS	FS	PbO **
Alternate	D ₂ O	D ₂ O	Li ₂ O	FS	FS	PbO **
Backup	Li	Li	Li	V	V	NR ***

* Austenitic stainless steel is an alternate selection for the first-wall structure and both austenitic stainless steel and titanium alloys are possible alternatives for the blanket structure.

** If required.

*** Not required.

is retained as the breeding material and ferritic steel as the first-wall/blanket structure. Table 3.2-9 summarizes the favorable and unfavorable features of the alternate blanket system.

Liquid lithium, which can be used as both coolant and breeder, provides a unique blanket option. However, the inherent safety of this system has been questioned and maintenance-related problems have been identified. An intermediate coolant loop may also be desirable. However, because the thermal-hydraulics, neutronics, radiation behavior, and design simplicity of this system are generally regarded as superior to other blanket materials options, this system is suggested as a backup. Selected vanadium alloys, which are proposed for the structural material, are believed to be the most resistant to radiation damage of the candidate structural materials, produce low long-term activation products, are compatible with lithium, and possess adequate elevated temperature mechanical properties. Long blanket lifetime may be attainable since the liquid lithium is not sensitive to radiation damage and tritium-release from the liquid also does not present a problem. Table 3.2-10 summarizes the favorable and unfavorable characteristics of the lithium blanket. The key problem of this option is to alleviate the safety concerns by appropriate design.

Table 3.2-8. Features of Helium-Cooled Blanket with Water-Cooled First Wall

Coolant: First Wall - D₂O
Blanket - Helium

Breeding Material: Li₂O

Structure: Ferritic steel

Concept: Pressurized D₂O coolant for first wall and inner blanket with helium coolant for ferritic steel-clad Li₂O outer blanket. Separate low-pressure helium purge stream for tritium recovery.

Unfavorable	Favorable	Characteristic
		<u>Safety</u>
X	X	Low stored chemical energy High pressure coolant release
		<u>Design</u>
	X	High-neutron wall load capability (D ₂ O)
	X	Small manifold requirement for first wall
X		High-pressure first-wall coolant
	X	Water liquid at room temperature
	X	Relatively easy cleanup of leak
	X	Low pumping power for first wall
X		Two-coolant system more complex
X		High-pressure blanket coolant
X		Helium requires large manifolding
X		Low thermal conductivity of Li ₂ O requires close coolant channel spacing
X		Requires low density, thick blanket
	X	No phase change in helium coolant
	X	Intermediate coolant loop not required
		<u>Neutronics</u>
	X	Adequate breeding capability without multiplier
X		Helium creates shielding problems
X		Long life activation products (Mo)
		<u>Thermal Hydraulics</u>
	X	D ₂ O excellent first-wall coolant (low-structure temperature)
X		Helium coolant requires relatively high-structure temperature
X		Poor conductivity of Li ₂ O increases "hot spot" problem
X		He requires high velocity to maintain low heat transfer coefficient

Table 3.2-8. Features of Helium-Cooled Blanket with Water-Cooled First Wall
(Continued)

Unfavorable	Favorable	Characteristic
		<u>Compatibility</u>
X	X	Good first-wall coolant compatibility
	X	Significant radioactive mass transfer from first wall
	X	Structure is compatible with hydrogen
	X	Coolant/breeder not highly reactive
X	X	Breeder/structure compatible at operating temperature
	X	Release of LiOT from Li ₂ O very corrosive to structure
	X	Coolant/structure compatible at operating temperature
		<u>Radiation Effects</u>
X		Radiation raises DBTT of ferritic steel above room temperature
	X	Ferritic steel exhibits low swelling and good high temperature ductility
X		Solid Li ₂ O susceptible to swelling, densification and radiation-induced sintering
X		Burnup of Li ₂ O
		<u>Mechanical Properties</u>
	X	Ferritic steel maintains adequate strength at D ₂ O temperatures
X		Helium outlet temperature is limited by mechanical properties of ferritic steel
X		High-pressure coolants increase stress requirements
X		Large helium duct sizes increase stress problems
		<u>Tritium Recovery</u>
X		Separate tritium processing fluid required
X		Tritium release from Li ₂ O is limited to specified temperature range
X		Tritium release from Li ₂ O partially in form of LiOT at high temperature, high fluence conditions
X		Radiation-induced trapping and sintering further restrict tritium release
X		Tritium inventory in structure relatively high
X		Tritium removal from D ₂ O requires extra processing equipment and costs
	X	Deuterium permeation through first-wall structure is not serious to plasma performance
		<u>Fabrication</u>
X	X	Ferritic steel requires post-weld heat treatment
	X	Ferritic steel can be welded in uncontrolled atmosphere

Table 3.2-9. Features of Water-Cooled (D₂O) Blanket

Coolant:	Pressurized water (D ₂ O)
Breeding Material:	Li ₂ O
Structure:	Ferritic steel
Concept:	Pressurized D ₂ O coolant for first wall and blanket with Li ₂ O breeder and ferritic steel structure and cladding. Separate low-pressure helium purge stream for tritium recovery.

Unfavorable	Favorable	Characteristic
		<u>Safety</u>
X		Water interacts with Li ₂ O (no H)
X		High pressure coolant release
		<u>Design</u>
	X	High neutron wall-load capability
	X	Small manifold requirement
X		High pressure coolant
	X	Water liquid at room temperature
	X	Relatively easy cleanup of leak
	X	Low pumping power
X		Low thermal conductivity of Li ₂ O requires close coolant channel spacing
		<u>Neutronics</u>
	X	Adequate breeding capability without multiplier
X		Long-lived activation products (Mo)
		<u>Thermal Hydraulics</u>
	X	D ₂ O excellent first-wall coolant (low structure temperature)
X		Low thermal conductivity of Li ₂ O increases "hot spot" problem
X		Limited energy conversion efficiency
	X	No intermediate coolant loop required
		<u>Compatibility</u>
	X	Good coolant/structure compatibility
X		Significant radioactive mass transfer
	X	Structure is compatible with hydrogen
X		Coolant/breeder reacts exothermically to form LiOH
	X	Breeder/structure compatible
X		Release of LiOT from Li ₂ O very corrosive to structure

Table 3.2-9. Features of Water-Cooled (D₂O) Blanket (Continued)

Unfavorable	Favorable	Characteristic
		<u>Radiation Effects</u>
X		Radiation raises DBTT of ferritic steel above room temperature
	X	Ferritic steel exhibits low swelling and good high temperature ductility
X		Solid Li ₂ O susceptible to swelling, densification and radiation-induced sintering
X		Burnup of Li ₂ O
		<u>Mechanical Properties</u>
	X	Ferritic steel maintains adequate strength at D ₂ O temperatures
	X	Small duct size reduces stress requirements
X		High-pressure coolant increases stress requirements
		<u>Tritium Recovery</u>
X		Separate tritium processing fluid required
X		Tritium release from Li ₂ O limited to specified temperature range
X		Tritium release from Li ₂ O partially in form of LiOT at high temperature, high fluence conditions
X		Radiation-induced trapping and sintering further restrict tritium release from Li ₂ O
X		Tritium inventory of structure may be high
X		Tritium recovery from D ₂ O requires extra processing equipment and costs
	X	Deuterium permeation through first wall is not serious to plasma performance
		<u>Fabrication</u>
X		Ferritic steel requires post-weld heat treatment
	X	Ferritic steel can be welded in uncontrolled atmosphere

Table 3.2-10. Features of Lithium-Cooled Blanket Concept

Coolant:	Lithium
Breeding Material:	Lithium
Structure:	Vanadium alloy (ferritic steel)
Concept:	Lithium circulates and serves dual-purpose breeder/coolant. Circulating lithium also serves as tritium-processing fluid.

Unfavorable	Favorable	Characteristic
		<u>Safety</u>
X		Lithium reacts vigorously with water (H)
X		Lithium reacts with air and concrete (no H)
		<u>Design</u>
	X	Combined breeder/coolant simplifies design
	X	Two-component system (V-Li) minimizes materials compatibility problems
	X	Low structural fraction requirement
	X	High-neutron wall load capability
	X	Small manifold requirements
	X	Low-pressure coolant
X		Lithium is solid below 180°C
X		MHD effects in flowing lithium
X		Vanadium has high permeability for hydrogen
X		Vanadium will oxidize in air at elevated temperature
	X	Low-pumping power requirement for coolant
X		Disruptive magnetic field forces
X		Spill/cleanup difficulty
X		Intermediate heat-transfer loop may be required
		<u>Neutronics</u>
	X	Good breeding capability
	X	Low (long-life) activation
	X	Good shielding characteristics (no duct leakage)
		<u>Thermal Hydraulics</u>
	X	Excellent heat transfer fluid
	X	High thermal efficiency
	X	Minimal "hot spot" problem
		<u>Compatibility</u>
	X	Good structure/coolant compatibility
	X	Structure compatible with hydrogen
	X	Coolant/breeder compatibility not an issue

Table 3.2-10. Features of Lithium-Cooled Blanket Concept (Continued)

Unfavorable	Favorable	Characteristic
		<u>Radiation Effects</u>
	X	Liquid lithium not sensitive to radiation damage
	X	Vanadium alloy highly radiation-damage resistant
		<u>Mechanical Properties</u>
	X	Vanadium provides adequate strength $\sim 100^{\circ}\text{C}$ above proposed operating temperature
	X	Low-pressure coolant reduces stress requirement
	X	Light-weight blanket materials reduce load
	X	Vanadium gives low-thermal stress factor
		<u>Tritium Recovery</u>
	X	Tritium-processing methods demonstrated
	X	Recirculating coolant provides tritium process fluid
	X	Tritium release not affected by radiation
		<u>Fabrication</u>
X		Difficult welding - high purity environment

3.2.3 Neutronics

Tritium breeding and induced activation are important neutronics considerations for STARFIRE. The feasibility of adequate tritium breeding in an actual engineering design must be carefully assessed, especially for solid breeders. This is the subject of Section 3.2.3.1. The characteristics of induced radioactivity in various structural and breeding materials are discussed in Section 3.2.3.2.

3.2.3.1 Tritium Breeding

A wide variety of blanket options have been considered for the STARFIRE reactor. The primary neutronics considerations are to achieve an adequate tritium breeding ratio and to maximize the energy production. The former is a requirement that must be satisfied. Enhancing the energy multiplication in the blanket is economically beneficial.

Previous studies have clearly shown that blankets employing liquid lithium as the breeding medium have an excellent tritium production capability sufficient to achieve an adequate tritium breeding ratio with a comfortable margin

to offset uncertainties in design and nuclear data. The tritium breeding potential in lithium compounds is lower because the additional elements will: a) increase the effective slowing down power; thus causing a reduction in the ${}^7\text{Li}(n,n'\alpha)t$ reaction rate, and b) absorb some of the neutrons that otherwise would be available for inducing the tritium-producing reactions in ${}^6\text{Li}$ and ${}^7\text{Li}$. It should be noted, however, that maximizing the tritium breeding ratio is not a goal for reactors in a mature fusion economy. For a doubling time of ~ 7 years with a low tritium inventory of ~ 3 kg as in the present STARFIRE design, the tritium breeding ratio required is ~ 1.01 . A higher breeding ratio will result in more tritium than can be used, which is not desirable.

The tritium breeding ratio is strongly influenced by the amount of the structural material in the blanket. Furthermore, it is sensitive to the heterogeneity of the system, the presence of penetrations, the coolant choice and the spatial distribution of the plasma neutron source. Therefore, the exact tritium breeding ratio can be determined only from three-dimensional calculations that incorporate all the geometrical details of the system. This makes the assessment of the tritium breeding ratio for different breeder materials difficult during the scoping stage of the reactor study when reliance has to be made on one-dimensional neutronics calculations.

Table 3.2-11 shows the blanket design options examined in the neutronics survey for STARFIRE. Liquid lithium and Li_7Pb_2 have excellent tritium production capability and achieving the required breeding ratio can be assured for a wide range of design conditions and the variety of structural materials considered in this study. For the lithium compounds in which the number of non-lithium atoms is much larger than the number of lithium atoms, the required tritium breeding ratio cannot be realized without the use of a neutron multiplier. Beryllium is the best neutron multiplier as it substantially enhances both the tritium breeding and energy multiplication. Unfortunately, beryllium is resource-limited. Lead is a good neutron multiplier. Adequate tritium breeding ratios appear to be obtainable with Li_2SiO_3 and LiAlO_2 provided that large amounts of lead (in the oxide form for reasons discussed earlier) are used.

Table 3.2-11. Blanket Design Options Used for STARFIRE Neutronic Study

Breeder	Neutron Multiplier	Structural Material	Coolant
Li_2O	none	Ferritic, titanium	He, D_2O
Li_2O	PbO , BeO , Pb, Be, Zr	Ferritic, titanium	He, D_2O
Li_2SiO_3	PbO , Pb, Zr	Ferritic, titanium	D_2O , He
LiAlO_2	PbO , Pb, Zr	Ferritic, titanium	D_2O , He
Li_7Pb_2	none	Ferritic, titanium	D_2O , He
Li	none	Vanadium, ferritic	Li, He

One of the primary reasons for selecting Li_2O as the breeding material for the STARFIRE reference design is the finding that it is the only suitable solid breeder that does not require a neutron multiplier. Table 3.2-12 shows the tritium breeding ratio (T) obtainable with Li_2O in two designs that differ in the type of blanket coolants. In design A, the blanket coolant is helium, but in design B it is heavy water. These results are based on 1-D calculations in which the blanket material composition is homogenized as shown in the table with the blanket completely encircling the first wall. In this idealized system, the breeding ratio is in the range 1.2-1.35 depending on the amount of structure and the type of coolant. Therefore, there is ~ 20% breeding margin (above the required $T = 1.01$) to compensate for the effects not included in the idealized calculations. One of these effects is the presence of major penetrations that have to be accommodated in the blanket. The present estimate is that with proper design choices this effect can be kept to < 5%. Another effect is the heterogeneity. This is difficult to estimate without detailed and expensive 3-D calculations but for the relatively simple reference design the effect of the heterogeneity on T can be kept to a few percent. Depending on the design, this effect may result in an increase or a decrease in T.

To summarize, it appears that an adequate tritium breeding ratio can be obtained with Li_2O without the use of neutron multiplier. Three-dimensional neutron transport calculations are presently underway to accurately determine the breeding ratio for the reference design. From the overall reactor economics standpoint, there is a strong incentive to reduce the blanket/shield thickness on the inner side of the torus (see Sec. 2.1). One important means

Table 3.2-12. Tritium Breeding Ratio For Two Blanket Options (a,b)

Design A

First Wall: 50% ferritic steel + 50% D₂O (1 cm)
 Blanket: Li₂O (natural Li) + ferritic + 30% He
 Reflector: 85% C + 5% ferritic + 10% He

	Volume % of Ferritic	T ₆	T ₇	T = T ₆ + T ₇
Case 1	10	0.93	0.34	1.27
Case 2	15	0.93	0.29	1.22
Case 3	20	0.92	0.25	1.17

Design B

First Wall: 50% ferritic steel + 50% D₂O (1 cm)
 Blanket: Li₂O (natural) + ferritic + D₂O
 Reflector: 90% C + 5% ferritic + 5% D₂O

	Volume %			T ₆	T ₇	T = T ₆ + T ₇
	Breeder	Coolant	Structure			
Case 4	90	5	5	0.94	0.41	1.35
Case 5	85	10	5	0.95	0.39	1.34
Case 6	85	5	10	0.94	0.37	1.31
Case 7	80	10	10	0.95	0.35	1.30

^a Results based on 1-D calculations with homogenized and uniform blanket completely encircling the first wall.

^b All percentages are volume percentages.

of achieving this is to replace part or all of the tritium breeding medium in the inner blanket with more efficient radiation attenuators. The trade-offs involved here are being investigated.

3.2.3.2 Reactor Activation

General Remarks

The requirement of achieving an environmentally acceptable reactor concept must be taken into account early in the design effort. The major

safety and environmental impacts of fusion power reactors, as envisioned today, are associated with the inventories of radioactive tritium and neutron induced activation of reactor components. In order to incorporate such safety and radiological issues into the design considerations, a careful consideration has been devoted to a prudent selection of breeding material, coolant and structural material.

The analysis in this section is focused on the effects of the following:

1. structural materials,
2. breeding materials,
3. coolant materials, and
4. neutron multiplier.

The analysis emphasizes the potential problems associated with each candidate blanket/shield system from the reactor activation standpoint. The candidate materials investigated in the present study are as follows:

1. Structural materials*	type 316 stainless steel (SS) ferritic stainless steel (FS) V-15Cr-5Ti alloy (V15Cr5Ti) Ti-4Al-2.5V-8Sn-0.5Si alloy (Ti4381)
2. Breeding materials	Li ₂ O Li ₇ Pb ₂ liquid Li LiAlO ₂ Li ₂ SiO ₃
3. Coolant materials	helium water liquid Li
4. Neutron multiplier	PbO

* See Table 3.2-13 for compositions.

The computation for the radioactivity and radioactivity-related parameters such as biological hazard potential (BHP), and decay β/γ spectrum has been performed by using the RACC⁽¹⁾ code in conjunction with the RACCDLIB/RACCLIB⁽²⁾ data libraries.

Table 3.2-13. Structural Material Composition

Element	Type 316 Stainless Steel		Ferritic Stainless Steel				V-15 Cr-5 Ti		Ti-4 Al-2.5 V-8 Sn-0.5 Si (Ti 4381)	
	wt-%	atom/b-cm	Type X		Type CE		wt-%	atom/b-cm	wt-%	atom/b-cm
			wt-%	atom/b-cm	wt-%	atom/b-cm				
C	0.058	2.286(-4)	0.09	3.478(-4)	0.080	3.089(-4)	0.02	6.118(-5)	0.01	2.267(-5)
N	0.007	2.366(-5)	0.049	1.621(-4)			0.05	1.311(-4)	0.008	1.555(-5)
O			0.01	2.898(-5)			0.05	1.148(-4)	0.065	1.106(-4)
Al			0.01	1.720(-5)			0.004	5.447(-6)	4.0	4.038(-3)
Si	0.460	7.752(-4)	0.08	1.320(-4)			0.03	3.924(-5)	0.5	4.848(-4)
P	0.026	3.974(-5)	0.014	2.100(-5)			0.01	1.186(-5)		
S	0.011	1.624(-5)	0.013	1.878(-5)						
Ti	0.040	3.953(-5)	0.09	8.713(-5)			5.00	2.835(-3)	84.9	4.825(-2)
V			0.200	1.820(-4)	0.150	1.365(-4)	79.794	5.754(-2)	2.5	1.336(-3)
Cr	16.70	1.520(-2)	8.500	7.581(-3)	9.000	8.025(-3)	15.00	1.060(-2)	0.01	5.236(-6)
Mn	1.430	1.232(-3)	0.28	2.363(-4)					0.0025	1.239(-6)
Fe	64.44	5.462(-2)	88.85	7.377(-2)	89.17	7.405(-2)	0.01	6.578(-6)	0.02	9.752(-6)
Co	0.030	2.410(-5)								
Ni	13.90	1.121(-2)	0.08	6.319(-5)			0.001	6.258(-7)	0.005	2.319(-6)
Cu	0.060	4.470(-5)	0.09	6.568(-5)					0.01	4.286(-6)
Ga							0.01	5.270(-6)		
Zr			0.001	5.281(-7)						
Nb			0.11	5.490(-5)	0.100	4.992(-5)	0.0025	4.996(-7)		
Mo	2.840	1.401(-3)	0.05	2.417(-5)	1.000	4.834(-4)	0.008	3.164(-6)	0.005	1.419(-6)
Sn									3.0	1.835(-3)
Ta							0.003	1.149(-6)		
W			1.480	3.733(-4)	0.500	1.261(-4)	0.0075	1.523(-7)		
Density (g/cc)	7.86		7.70		7.70		6.10		4.52	

Effect of Structural Materials on Reactor Activation

Figure 3.2-1 shows a comparison of the candidate structural materials in terms of biological hazard on inhalation (BHP air) as a function of time after reactor shutdown following a two year reactor operation. Over the time period from 5 years to 50 years after shutdown, the two different ferritic steel systems (FS-CE and FS-X) show a factor of 10-50 lower BHP than the 316 stainless steel alloy. Over the same time period, the BHP in the Ti4381 system is further reduced by approximately one to two orders of magnitude from those for the FS systems. Because of the similarity of the isotope contents in the Ti4381 and V15Cr5Ti alloys, the general trend of BHP variation in the both systems is almost identical, keeping about a factor of 10 difference in their absolute magnitudes over the entire time span under consideration.

Figure 3.2-1. Effect of structural materials on biological hazard potential. The results are based on the following system: first wall: 10 mm structural material; blanket: 0.6 m (90% Li + 10% structure); shield: 0.9 m (50% B₄C + 50% structure).

Beyond 50 years after reactor shutdown, the BHP levels are ordered as follows:

$$\text{BHP(SS)} \approx 10 \text{ BHP(FS-X)}$$

$$\text{BHP(FS-X)} \approx 10 \text{ BHP(Ti4381)}$$

$$\text{BHP(Ti4381)} \approx 10 \text{ BHP(V15Cr5Ti)}$$

Due to a relatively high molybdenum content (1 w/o), the FS-CE system shows a constant level of BHP ($\sim 1.5 \text{ km}^3/\text{MWth}$) beyond 50 years after shutdown.

Figure 3.2-2 shows the isotopic contribution to the BHP in the FS-X system.

All of the candidate structure systems show more or less the same trend of isotopic contributions to the total BHP's over 5-50 yr after shutdown. The most dominant radioactive isotopes in this time range are Co-60 (5.3 yr, β^-) and Fe-55 (2.7 yr, EC). The Co-60 is induced directly or indirectly from a series of nickel isotopes (through reactions such as Ni-60(n,p), Ni-61(n,d), Ni-62(n,t) and Ni-60(n,p) Co-60 m $\rightarrow$ Co-60). On the other hand, the largest contributions to Fe-55 generation are made through the Fe-54(n, γ), Fe-56(n,2n)

Figure 3.2-2. Isotopic contribution to BHP in air for Li blanket system with ferritic steel type X as the structural material.

and Ni-58(n, α) reactions. This observation provides a qualitative explanation about the low BHP levels in the time period in question for the Ti4381 and V15Cr5Ti systems in which both contents of Ni (order of 0.001 w/o) and Fe (order of 0.01 w/o) elements are extremely low compared to the other systems. Beyond 50 years after shutdown, the BHP's are determined largely by the impurity levels, in particular, by Nb and Mo that lead to the formation of Nb-93 m (16.6 yr. IT), Nb-94 (20,000 yr, β^-) and Mo-93 (3,000 yr, EC). The only exceptions in which the primary structural elements represent the major sources for long-lived radioactive isotopes, are: 1) Al-26 (7.2×10^5 yr, β^+ /EC) induced by the Al-26 (n,2n) reaction in Ti4381; and 2) Ni-63 (100 yr, β^-) induced by the Ni-62 (n, γ) and Ni-64 (n,2n) reactions in 316 stainless steel.

Effect of Breeding Materials on Reactor Activation

A comparison of BHP's in each reactor component for the candidate breeder systems has been made. The structural material is assumed to be ferritic stainless steel (FS-X) throughout the reactor systems. With regard to the coolant, a helium gas is used in all cases except for the liquid lithium blanket which is also cooled by the lithium fluid. It is found that the BHP's in the whole system, as well as in each component, do not vary to any appreciable degree from one breeder system to another. The total BHP's for all the systems are bracketed within a maximum difference of a factor of ~ 2 at each time point over the time span of 0-100 yr. However, it should be noted that the results obtained here are very dependent upon the structural material chosen. In the case of FS-X, the effect of spectrum change (resulting from use of a different breeder material and/or PbO multiplier) on the BHP's tends to be less significant simply due to the large BHP of the structural material itself.

Effect of Coolant Materials and PbO Multiplier on Reactor Activation

Figure 3.2-3 illustrates the impact of the use of PbO neutron multiplier on the BHP air for two blanket designs currently being studied; i.e.,

1. water-cooled $\text{Li}_2\text{SiO}_3/\text{PbO}$ blanket, and
2. helium-cooled $\text{Li}_2\text{SiO}_3/\text{PbO}$ blanket

both based on the breeder pin concept in which the clad Li_2SiO_3 breeder pins are embedded in a bulk of the PbO neutron multiplier. Also included in Fig. 3.2-3 is the liquid lithium breeder/coolant system studied earlier

Figure 3.2-3. Effect of coolant materials

(see Fig. 3.2-1) for the sake of comparisons. The liquid lithium case is representative of breeding blankets with no neutron multiplier (e.g. Li_2O blanket). The structural material is assumed to be the Ti-4Al-2.5V-8Sn-0.5Si (Ti4381) alloy.

As shown in Fig. 3.2-3, the liquid lithium system yields a slightly higher BHP up to ~ 1 yr after shutdown than those for the helium- or water-cooled solid breeder systems. Such a high BHP in the liquid lithium system stems from the fact that the hard neutron spectrum inherent in the lithium blanket design enhances the several threshold reactions (such as V-50 ($n, n'\alpha$), Ti-49 ($n, n'\alpha$), Ti-48 (n, α), Ti-47 (n, d), Ti-46 (n, p)) leading to the major radioactive isotope generations of Ca-45 (165 d, β^-) and Sc-46 (83.8 d, β^-). Beyond approximately two years after shutdown, both of the solid breeder systems exhibit higher levels of BHP than that of the liquid lithium system. The difference is magnified, particularly around ~ 5 yr and beyond 50 yr after shutdown, reflecting the effect of the PbO multiplier in the solid breeder blankets. Isotope Ti-204 (3.77 yr, β^-/EC), which is produced through the

threshold Pb-204 (n,p) and Pb-206 (n,t) reactions in the multiplier region, predominates in the time interval of 5-30 yr. This observation is consistent with the fact that the helium-cooled blanket shows a BHP level persistently higher than that of the water-cooled system (although the difference is not appreciable) in the time range in question. Beyond 50 yr, in particular, beyond 100 yr after shutdown isotope Pb-205 (1.4×10^7 yr, EC) induced by the Pb-206 (n,2n) reaction, which also takes place in the multiplier region, represents more than 90% of the total BHP in the solid breeder systems. The activation level of Al-26, which is the most dominant BHP isotope on a long-time scale in the case of the Ti4381 structured Li system, now represents only a small portion of the total BHP in the case of the solid breeder systems.

Due to the lack of data in the current data library, some important radioactive isotopes were not included in the present analysis. One of such isotopes is Bi-207 (38 yr, EC/ β^+) which is produced by the neutron-PbO multiplier interactions such as Pb-207 (n,p), Pb-208 (n,d) and Pb-208 (n,n'p). The electron capture decay of Bi-207 is accompanied by three major gamma rays of 0.57 MeV, 0.90 MeV and 1.4 MeV. The effect of inclusion of such important isotopes remains to be examined.

3.2.4 Tritium and Safety Considerations for the First Wall and Blanket

A comprehensive study of various structure/breeder/coolant systems has recently been completed wherein details of experiments investigating potential fusion breeders and coolants are discussed,⁽³⁾ and hence, will not be reiterated here. This section addresses the key conclusions of assessments of tritium and safety issues pertaining to three specific areas, namely:

1. The implications of water (or D₂O) as a first wall coolant;
2. Assessment of tritium recovery and temperature limits for candidate breeder materials; and
3. Coolant/breeder/tritium safety considerations.

3.2.4.1 Implications of the Choice of Water (or D₂O) as a First-Wall Coolant

During the course of reactor operation, significant amounts of tritium will pass from the plasma chamber into the first wall coolant (water). The tritium is essentially all converted to HTO, in which form permeation rates

are very small ($\ll 0.1$ Ci/day) and tritium is lost only to the extent that the coolant water itself is physically lost. Thus, water is a sink for tritium. Permeation rates for tritium passing from the plasma chamber through the ferritic steel tube wall into the first wall coolant were calculated (Table 3.2-14). First wall temperatures of 300 and 500°C were assumed for the calculation. The driving pressure was assumed to be 0.13 Pa (10^{-3} torr) and the attainable barrier effectiveness was assumed to be 400-X. Comparison of these results with other materials of construction show that permeation rates for stainless steels will be an order of magnitude lower, while more permeable materials, e.g., Ti, V, and Nb alloys, will be about two orders of magnitude higher. Steady-state levels (Table 3.2-14) are achieved when β -decay equals about 20 times the annual inleakage rate, assuming that losses are insignificant and that tritium is not actively removed from the water.

Table 3.2-14. Summary of Tritium Loss Calculations for the Tube Bank First Wall Concept

Material = Ferritic Steel

Area = 730 m²

Thickness = 3 mm (3×10^{-3} m)

Average Driving Pressure = 10^{-4} kPa ($\sim 10^{-3}$ torr)

Parameter	<u>With Barrier^c</u>		<u>Without Barrier</u>	
	300°C	500°C	300°C	500°C
Permeability, Ci/day·m·kPa	10^{-3}	10^{-2}	0.4	4
Permeation Rate, Ci/day	2.5	25	1×10^3	1×10^4
Permeation Rate, Ci/year	10^3	10^4	4×10^5	4×10^6
Steady State Level, ^a Ci	2×10^4	2×10^5	8×10^6	8×10^7
Steady State Concentration, ^b Ci/l	0.2	2	80	800

^a Level at which decay rate equals migration rate

^b Assuming 10^5 liter of coolant. Comparable levels for heavy water moderated fission reactors are ~ 80 Ci/l in the moderator and ~ 4 Ci/l in the primary coolant.

^c Assuming a permeation barrier effectiveness of 400-X.

For conditions anticipated in systems with a water-cooled first wall and/or blanket, tritium levels are estimated to be 2 to 100 Ci/l. As a point of reference, 20 Ci/l is typical of the level in the moderator of a heavy water reactor (HWR) after many years of operation, however, some control is implemented to maintain this level. This level gives H/T ratios on the order of 10^5 in the water and requires an enrichment of at least six orders of magnitude before one can consider returning the tritium to the fuel cycle.

The use of D_2O as a coolant offers a significant advantage over H_2O because deuterium is a fuel, and therefore, isotopic enrichment requirements are relaxed. For example, the tritium could be recovered in a 99% deuterium mixture, requiring enrichment factors of $\sim 10^3$, which is three orders of magnitude lower than required for light water. Separation factors will be somewhat lower for D_2O , but this is a comparatively small effect. Also, for STARFIRE, 2000 Ci/yr will be generated in the D_2O coolant from neutron activation of the deuterium. This amount is less than that for inleakage from the plasma. Although hydrogen permeation from the coolant through the first wall could affect the plasma performance, deuterium permeation will present no problem. All factors considered, it appears that D_2O is much preferred over H_2O as a coolant.

The amount of the tritium in the cooling water circuit that will leak to the environment is difficult to ascertain. At the Pickering CANDU HWR [2160 MW(e)] $\sim 0.3\%$ of the annual tritium produced is released (10^7 Ci/yr produced versus 3×10^4 Ci/yr released). It is worthy to note that ~ 66 of the ~ 81 Ci/d released from the Pickering plant goes up the stack while only 16 Ci/d is lost via the cooling water. It has generally been assumed that the cooling water loss would be the only significant tritium release from a fusion plant. Continuous scrubbing of all stack effluents to remove tritium may be required.

The amount of tritiated water to be disposed of cannot be determined at this time. It is estimated to be from 10^3 to 10^5 liters. Since water is a sink for tritium, high levels of tritium are not expected in the structural materials. Water leakage during maintenance or accidents is the major concern.

Cross leakage at the steam generator that could couple the first wall coolant circuit to the power generator systems must be considered. Tritium contamination in decommissioned structural materials should remain relatively well fixed providing the stored waste is kept at or near ambient temperature and away from water. In cases where used reactor materials are stored for recycle, the tritium present in these materials will decay away faster than most of the other radio-nuclides that will be present, i.e., 100 years of internment would get rid of virtually all of the tritium.

The Canadians are developing an enrichment technology that they believe will ultimately allow them to recover nearly pure tritium at around 50¢/Ci. The process is not simple and it is believed that the ultimate cost per Ci could be considerably higher. However, this type of technology is vital to fusion, to insure a maximum recycle of tritium to the fueling system with only minimal tritium leakage and disposal problems.

There are three potential operating scenarios for a water coolant circuit on STARFIRE: (i) Allow tritium to build up until the decay rate equals the migration rate, providing the steady-state level is ~ 20 Ci/l; (ii) provide sufficient in plant enrichment capacity to maintain the level at ~ 20 Ci/l (if the migration rate is too high) and dispose of the tritium-rich fraction; and (iii) provide sufficient in-plant enrichment to recover all migrating tritium for recycle to the reactor fueling systems. Clearly, the last of these options is to be preferred.

3.2.4.2 Assessment of Tritium Recovery and Temperature Limits for Candidate Breeder Materials

For solid breeders, tritium is recovered from a helium purge stream by sorption on molecular sieves. The tritium pressure in the purge stream is $\sim 10^{-4}$ torr and the amount of tritium in the He is ~ 1 g. The tritium inventory in the solid breeder will be determined by diffusion rates. Thermodynamic considerations will have only second-order effects upon the inventory in a solid breeder.

In a fusion blanket, the concentration in the solid breeder will build up until the rate of diffusion out of the solid is equal to the tritium

generation rate, and steady state is achieved. At steady state and for infinite slab geometry, this is described by the relationship:

$$I = \frac{1}{3} \dot{I} \frac{r^2}{D} \quad (1)$$

where I = tritium inventory,
 $\dot{I}$ = tritium generation rate,
 r = slab half-width, and
 D = diffusivity.

In typical experiments,⁽⁴⁾ when diffusion rates are measured by first irradiating the sample and then heating to remove tritium, the diffusion kinetics are not steady state. The diffusivity D is conservatively estimated, using the steady-state equation for spheres

$$D = \frac{r^2}{15 \tau} \quad (2)$$

where τ = mean residence time for tritium in solid = time to extract 87.4% of the tritium.

In the experiments, τ can be readily measured; but, it is much more difficult to determine r , and therefore, D . However, assuming the r in the experiments is representative of a fusion blanket, equations (2) and (3) can be combined eliminating r and D , giving

$$I = 5 \dot{I} \tau \quad (3)$$

For a 3000 MWe reactor (steady state) at 20 MeV/fusion and a breeding ratio of 1.20, the tritium generation rate ($\dot{I}$) is 5.60×10^{-3} g/s. If τ is expressed in hours, then

$$I(g) = 5.560 \times 10^{-3} \cdot 3600 \cdot \tau(hr) = 100 \cdot \tau(hr)$$

Using this relationship, the calculated tritium inventories for candidate solid breeders are shown in Fig. 3.2-4. In view of the very strong

Figure 3.2-4. Predicted tritium inventory in candidate solid breeding materials (20 mesh) as a function of temperature after low-fluence irradiations.

temperature dependence and the fact that the inventory is extremely sensitive to the diffusion path length r ($I \propto r^2$), it is not surprising that some analyses conclude that solids will have very low tritium inventories, while others conclude that solids may have very high inventories.

The design solution to this dilemma is proper temperature control. The minimum temperature must be sufficiently high that tritium will diffuse out at an adequate rate, and the maximum temperature must not permit excessive agglomeration and sintering, which could increase the diffusion path length and thereby reduce diffusion rates. It is suggested that, as a design criterion, the inventory of any 5% of the blanket not exceed 500 g per GWth.

Therefore, not more than 5% of the blanket can be below the minimum temperatures indicated in Table 3.2-15 for each candidate breeding material. Therefore, a minimum temperature is established for each solid breeder (Fig. 3.2-4). The maximum temperature for ceramics may be determined by analogy to oxide fuels for fission reactors, wherein radiation enhanced sintering occurs at

Table 3.2-15. Temperature Limits (°C) for Solid Breeders

Breeder	m.p. °C	Thermal Effects		Radiation Effects	
		T _{min} ¹	T _{max} ²	T _{min} ³	T _{max} ⁴
Li ₂ O	1700 ⁵	410	1000	460	910
LiAlO ₂	1610	500	1000	550	850
Li ₂ SiO ₃	1200 ⁶	420	900	470	610
Li ₂ Si	760	480	550	530	420
LiAl	700	300	500	350	380
Li ₇ Pb ₂	726	320	530	370	390

¹ Diffusion limited.

² Thermal sintering.

³ Radiation-induced trapping (factor of 10 degradation in tritium release estimated).

⁴ Radiation enhanced sintering ($0.6 T_m$ for oxides and $0.67 T$ for other compounds).

⁵ Li₂O reportedly sublimes at temperatures below the melting point.

⁶ Li₂SiO₃ is a line compound with a solidus temperature of ~ 1030°C for slightly hypo- or hyperstoichiometric compositions.

~ 0.6 of the absolute melting point. For alloys, the radiation sintering temperature is estimated to be $2/3$ the absolute melting point. It should be pointed out that Wiswall⁽⁴⁾ performed prolonged heat treatments at 80 to 90% of the absolute melting point for several materials and found that, although the specific surface areas decreased substantially, the rates of tritium release increased significantly. On the other hand, he found that tritium release rates decreased with increasing fluence (up to 10^{20} n/cm²). The irradiation temperatures were close to room temperature, however, and it is not likely that radiation induced sintering occurred. The effect may be due to creation of radiation-damage-induced trapping sites. This phenomenon requires experimental investigation. The implication for STARFIRE is that the lower temperature may have to be increased (perhaps $\sim 50^\circ\text{C}$) or the inventory may increase by a factor of ~ 10 .

The above considerations are translated into temperature limits shown in Table 3.2-15. Provided the breeder is kept within the above temperature constraints, it is estimated that the blanket tritium inventory for the reference case Li_2O will be of the order of 1 kg. Tritium recovery from liquid lithium appears to be rather straightforward and there is confidence that levels of ~ 1.0 wppm or less can be achieved. This would correspond to a blanket inventory of 100 g.

3.2.4.3 Coolant/Breeder/Tritium Safety Considerations

In the absence of a detailed design, safety considerations cannot be quantitatively assessed. Therefore, generic safety considerations associated with blanket design are assessed in this section. Effects related to tritium migration into the first wall coolant have been discussed above.

Chemical reactivity of breeder or coolant is a significant concern. Helium coolant is not inherently reactive with air or breeder and structural materials, and it is the best choice from reactivity considerations. Water coolant has vigorous exothermic reactions with lithium and lithium alloys to the extent that these breeder/coolant combinations are not considered viable. Water coolant should have only modest reactions with Li_2O (or other oxides) at elevated temperatures. Cleanup of reaction products and corrosion of the structural materials by LiOH pose the major problems with this system. The reactivity of lithium with water, air, and concrete and the associated accident scenarios and cleanup problems are matters of significant concern.

Coolant pressure is a significant safety concern. Pressurized water and helium would both have pressures considerably higher than liquid lithium.

Tritium safety implications are concerned with tritium inventories and the chemical form of tritium. Because tritium is not a radiation hazard, but is an assimilation hazard, the more absorbing oxidized form (T_2O or HTO) has a significantly higher biological hazard than does T_2 . Coolant inventories will be lower (~ 1 g) for He or water, but the tritium will be in the oxidized form. In the liquid lithium case, the tritium inventory in the coolant will be the blanket inventory, i.e., ~ 100 g.

3.2.5 Mechanical Design of First Wall/Blanket

First-wall/blanket mechanical design concepts are being developed for the selected reference and alternate materials options discussed in Sec. 3.2.2.5. This section discusses rationale for the selection of the design concepts for the reference helium-cooled and alternate water-cooled blanket choices, presents the concepts and discusses their relative merits. The design concept for the backup lithium-cooled blanket is also described briefly.

The reference and alternate design concepts are presently being developed and analyzed further to assure that design requirements are satisfied in the areas of thermal hydraulics, power cycle efficiency, tritium breeding, tritium recovery, fabricability, and structural integrity, and to determine how well they satisfy the overall project objectives. The concepts will subsequently be compared and the final choice made for the reference first wall/blanket materials combination and mechanical design concept.

The overall approach to mechanical design of the first wall/blanket system is similar for both the reference and alternate materials options. The most important objectives for the designs were to:

- minimize probability/consequences of accidental contact of incompatible materials;
- maximize reliability of system and components, to increase availability;
- maximize maintainability by minimizing need for, and time required for, standard maintenance operations;

- minimize inboard wall and blanket thickness, to reduce reactor size and maximum magnetic field; and
- minimize outboard wall and blanket thickness, to minimize TF and EF coil size.

Selection of the mechanical design concept for the reference helium cooled blanket and the alternate water cooled blanket is strongly dependent on the individual and combined characteristics of the selected structural material, coolant and breeder. Of critical importance is assuring satisfactory adherence to minimum and maximum temperature constraints placed on the solid breeders (Sec. 3.2.4.2). For the helium cooled design, minimizing the blanket thickness, manifold and header size were also of major importance, in order to minimize required reactor size.

Both the reference and alternate designs use the concept of individual blanket modules of similar shape and size which are built up into first wall wedge-shaped blanket circumferential sectors. This approach has several advantages. It permits more accurate tailoring of the blanket to match neutron wall load values at specific positions around the plasma cross section. Accommodation of local discontinuities such as vacuum ports and rf ducts is simplified. Replacement of individual modules can be performed in the hot cell, while the reactor is returned to normal operation following installation of a replacement sector. The reactor is thus shut down only for the time needed to remove the sector containing a faulty first wall or blanket component and to install the replacement sector.

A water cooled first wall, mechanically and structurally separate from the blanket, was selected for both the reference and alternate concepts. First wall design life is expected to be significantly less than that of the blanket. The separate first wall concept permits simple replacement (in the hot cell) of a failed first wall panel without also requiring replacement of the more expensive and longer lived blanket module behind it. The use of water coolant for the first wall in the reference design permits accommodation of the selected neutron wall load (P_{nw}) value of 3.5 MW/m^2 with negligible pumping losses. The two additional disconnect/reconnect operations for the first wall water coolant headers, necessary for removal of a reference blanket sector, are not considered a major addition to the time required for that operation.

3.2.5.1 Reference Design

The reference first wall/blanket design features a helium cooled blanket. Lithium oxide (Li_2O) is used to breed tritium, which is removed continuously from the breeder through a helium purge gas system. The breeder is sealed to prevent contact with the helium coolant stream. Ferritic steel alloy is the baseline structural material for the blanket and the first wall. The first wall is water cooled and separate from the blanket.

Two mechanical design concepts are being considered for the reference blanket. In the first concept, the module walls are pressurized to the coolant static pressure. The solid breeder is held in sealed tubes, arranged in a staggered rod bank pattern, which are cooled by cross-flowing the helium over them. In the second concept, the helium coolant flows inside tubes each of which is surrounded by solid breeder throughout the module. The two concepts will be compared in the near future after further evaluation to determine which is superior from an overall reactor design standpoint. The remainder of this section discusses the two concepts, and the water cooled first wall common to both.

Pressurized Module Concept

The pressurized module has essentially the form of a slightly tapered parallelepiped, as shown in Fig. 3.2-5. Four of the sides are aligned to follow the two planes formed by the wedge-shaped blanket sector and two planes extending radially, approximately from the plasma center. The front face is semicircular and the back face is a shallow semiellipsoid. Module width (distance along sector chord) is adjusted to match sector width at any point. Module depth and thermal hydraulic parameters (breeder zone details and coolant mass flowrate) may be adjusted to match the actual neutron wall load value for any position in the blanket sector.

Each module has a single coolant inlet and two coolant outlets at the rear face, which connect to manifolds wrapping around the blanket sector. The manifolds connect to vertically aligned headers (large diameter pipes) at the top and bottom of the sector. The headers are attached mechanically to inlet and outlet feed pipes which remain in the reactor when a sector is removed.

Figure 3.2-5, Pressurized module concept.

Figure 3.2-5. (Continued) Pressurized module concept.

The helium coolant enters the module at the rear face and flows radially toward the nose (front face) through plenums formed by the double walled module sides. Helium enters the semicircular nose area and the flow is divided into channels, with flow direction alternating between adjacent channels. The helium then flows into a plenum region and is directed radially through the breeding region. The flow then enters a second plenum region and exits to the outlet manifolds.

The breeding zone of the blanket module contains circular breeder tubes, which extend across the full module width (toroidal direction). The tubes are in rows, with successive rows staggered to result in a triangular pitch. Each tube consists of a structural shell, Li_2O breeder, and end fittings for the purge gas system. The shell is sized for a differential pressure (collapse) equal to the coolant static pressure. The compacted lithium oxide breeder inside the shell has a small axial central hole. The helium purge gas is introduced into the cell through the fitting at one end, flows through the breeder center hole, and exits through the fitting at the other end. The helium (at ~ 1 atm pressure) permeates the breeder, and collects the tritium in the form of T_2O or LiOT with oxygen supplied through a low partial pressure of oxygen in the helium. The end fittings connect the tube end plates to the module side walls to complete the purge gas path.

The breeder tube diameter varies through the depth of the blanket, in order to maintain breeder temperatures within minimum and maximum temperature limits (Sec. 3.2.4.3) as volumetric heating rates decrease with depth through the blanket. The breeder temperatures are quite sensitive to changes in the heat conductance value ($h = 2000 \text{ W/m}^2\text{-}^\circ\text{K}$) assumed for the structure/breeder interface, and further work is needed to optimize breeder tube design details to assure that required ranges of conductance values can be reliably maintained throughout the blanket design life.

Heat generated within the breeder is conducted through the breeder to the tube structure surface. The helium coolant, flowing across the rows of tubes, removes this heat by convection. Thermal-hydraulic characteristics of the coolant flow across the breeder tubes are based on well-established empirical relationships for cross-flow tube bank heat exchangers. Spacing of the tubes is based on an equilateral pattern for adjacent tubes with 1.15 D spacing between tube centers (i.e., the gap between any two adjacent tubes is

equal to 0.15 times the tube diameter). This permits a relatively high packing fraction for the tubes which helps to reduce the blanket depth required to achieve acceptable breeding levels. Total pumping power required for the breeding zone only is ~ 0.5 MW (work) for the reactor which is negligible in comparison to the power required for the rest of the helium coolant system. Total pumping power for all 24 blanket sectors is estimated to be ~ 50 MW (work), or $\sim 1.3\%$ of total reactor thermal megawatts, with nearly all the losses occurring in the manifolds. This value includes all pressure losses occurring in the sectors between the inlet and outlet header disconnects. These relatively low values result from designing for minimum flow restrictions within the blanket modules and within the manifolds, and from the use of water instead of helium to cool the first wall and inboard (nonbreeding) blanket.

The pressurized module has been designed to make efficient use of structural material to keep the structural volume fraction as low as possible in order to enhance tritium breeding. In addition, nearly all the area of the module structural walls, which are subjected to a differential pressure approximately equal to the 750 psig coolant static pressure, is cooled by flowing helium to limit maximum wall temperature to $\sim 450^\circ\text{C}$ or less. The module sides are designed with outer and inner walls joined by ribs. This design is structurally more efficient (results in a lower percentage volume of structure) than a single pressurized wall. The use of two walls also provides a flow channel through which relatively cool helium from the inlet manifold flows radially toward the blanket nose, keeping the pressurized outer wall at $\sim 300^\circ\text{C}$ or less. The inner walls experience virtually no differential pressure. Tie rods span the long width direction of the module in a roughly square pattern with ~ 10 cm spacing. These rods react the pressure loads applied to opposite walls, further reducing the required wall thickness. In the short width direction, a combination of the tie rods and thin bulkheads is used to react pressure loads on opposite walls. The bulkheads also provide support for the long breeding cells by holding them in close-fitting holes to prevent any significant deflection of breeder tubes between bulkheads. The possibility of using the breeder tubes as structural members, to eliminate tie rods in the long width (tube axial) direction, will be investigated.

Coolant inlet temperature and outlet temperature for the pressurized module concept are 250°C and 500°C, respectively. A key feature of this concept is that it permits the maximum coolant temperature to be approximately equal to the maximum temperature of the module wall structure. This is important because it maximizes the power cycle efficiency for the helium coolant. The only pressurized structure in the module with a temperature nearly equal to coolant outlet temperature is the outlet manifolds at the rear of the blanket. All other structure in the module, with the exception of the breeder tubes near the rear of the blanket, will experience temperatures no greater than the coolant outlet temperature. The breeder tubes at the rear of the blanket are estimated to have a structural temperature approximately 30°C higher than the coolant, or ~ 530°C. Most of the structure will be at considerably lower temperatures, and will be sized for stresses resulting from thermal gradients and reactions to pressure loads.

The pressurized module concept for the reference blanket has been analyzed to determine a preliminary estimate for volume fractions, assuming both ferritic steel alloy and titanium alloy Ti-4381 (see Sec. 3.2.2.3) as structure. The table below shows the results.

	<u>Ferritic</u>	<u>Ti-4381</u>
% V Structure	14.9	11.1
% V Li ₂ O*	57.2	57.2
% V Void/He	27.9	31.7

* Volume required by breeder without accounting for voids.

The lower structural fraction for the Ti-4381 results from the higher allowable stresses compared to the ferritic steel. From the standpoint of design and fabricability the titanium alloy seems superior to the ferritic steel because (a) its ductile-to-brittle transition is significantly less than room temperature, (b) no preweld heat treatment is required, and (c) postweld heat treatment (other than annealing) is not required.

Pressurized Tube Concept

In the second concept for the reference helium cooled blanket, helium coolant is contained in tubes which are surrounded by the breeder (essentially

the reverse of the concept previously described). This concept is shown in Fig. 3.2-6. It is very similar in approach to the alternate water-cooled blanket described in the next section. The tubes are each in a U-shape. The tube end regions are aligned radially along the module ends, with the center region running lengthwise through the module. The ends of the tubes are manifolded so that each module has only one inlet and one outlet fitting for the coolant. These fittings are located on the flat back face of the module.

The outer wall of the module is sized to contain the ~ 1 atm static pressure of the helium purge gas. The wall is water cooled, and the construction is similar to that of the water cooled first wall. Active cooling is necessary because the breeder conductivity is too low to adequately conduct heat away from the outer wall to maintain a satisfactory maximum structural temperature.

The module walls enclose the breeder which fills the module and surrounds the tubing network. Tube spacing and tube inside diameter are varied through the blanket depth to achieve equal coolant temperature changes and equal coolant pressure drops for all the tubes. Preliminary thermal-hydraulics analysis indicates that tube inside diameters should vary from 1.27 cm at the front to 0.8 cm at the rear of the blanket. Approximately 173 tubes are required for the blanket. A narrow gap between the breeder and the tube outer surface has been conservatively assumed. Coolant inlet and outlet temperatures have been assumed to be 225° and 475°C, respectively. Outlet temperature was reduced from the 500°C assumed for the previous concept to keep the maximum tube temperature to $\sim 560^\circ\text{C}$. Pressure drops for the blanket have been estimated to be 16 to 18 psi, which results in pumping power losses of $\sim 2.2\%$ of total reactor thermal megawatts.

For both concepts, the inboard blanket is presently assumed to be water cooled and non-breeding, to minimize Δ_{BS}^1 (see Sec. 2.1). However, should neutronics analysis show that additional breeding on the inboard wall is needed for either concept, a modified breeding blanket design would be adopted for that region. The design would essentially consist of the first 40 to 60 cm depth of the standard breeding blanket concept. Blanket depth and manifolding diameter would be adjusted in this region to maintain the desired Δ_{BS}^1 and still achieve the necessary additional breeding and adequate heat recovery.

Figure 3.2-6. Blanket module cross section - helium cooled.

The pressurized tube concept has the relative advantage of simplicity compared to the pressurized module concept. The high pressure coolant is contained by tubes rather than by the irregular shape formed by the module walls. The circular shape is structurally more efficient for containing the high pressure. The reduction in the number of stress concentration regions could be expected to result in somewhat greater assurance against the occurrence of coolant leaks.

However, the pressurized tube concept has several disadvantages. First, the maximum temperature of each tube will be $\sim 75^{\circ}\text{C}$ higher than the coolant outlet temperature. The wall thickness of the tubes must, therefore, be sized to the maximum allowable structure temperature and the coolant outlet temperature determined accordingly. For the pressurized module concept, maximum structure temperature is $\sim 530^{\circ}\text{C}$ and occurs in the breeder tubes near the rear of the blanket. On the basis of preliminary analysis, given the inherent differences between maximum coolant temperature and maximum structure temperature for the two concepts, for any given arbitrary maximum structure temperature limit the pressurized module concept should permit coolant outlet temperatures 40°C to 50°C higher than for the pressurized tube concept. Second, the pumping power required for the blankets and manifolds together is estimated to be $\sim 3.0\%$ of total reactor thermal megawatts; this compares to $\sim 1.3\%$ for the pressurized module concept. The difference amounts to ~ 66 MW (work), and would significantly reduce net efficiency for the power cycle. Further work is required to reduce these pumping losses. The requirement for an actively cooled module wall is a significant added complexity and may possibly a significant reduction in blanket breeding capability. This requirement will be further investigated.

First Wall

The first wall concept for the reference design is essentially a water cooled flat panel. The basic concept is common to both the reference and alternate blanket concepts. Details of the design will be developed later in the study. For the present, the mechanical design concept used is that adopted previously in the ANL-EPR study.⁽⁵⁾ In this design, cooling is accomplished by circulating pressurized water in a network of channels that line the plasma side of the panel. The channels are formed by bonding pre-formed steel sheet to a second, flat steel sheet. Each panel section spans

several blanket modules, and has one inlet and one outlet coolant line which connect to manifolds at the rear of the blanket module. These manifolds connect all first wall panels and the water cooled inboard blanket within a single blanket sector. The manifolds terminate in headers in the vicinity of the helium blanket coolant headers, and are in turn attached mechanically to inlet and outlet feed pipes which remain in the reactor when the sector is removed.

3.2.5.2 Alternate Design

A pressurized water cooled blanket has been designed as an alternate to the reference helium cooled blanket and is currently being analyzed. The advantages of the water cooled blanket include very low pumping power requirements, smaller diameter coolant lines external to the blanket, and virtual elimination of coolant line neutron streaming effects. Some of the disadvantages of the water cooled blanket relative to the helium cooled blanket are a reduced gross power cycle efficiency and the consequences of an accident or leak involving the release of water coolant into the lithium oxide breeder.

The alternate water cooled blanket design is shown in Fig. 3.2-7. The coolant is heavy water, D_2O , contained in tubes arranged on a triangular pitch which penetrate lengthwise through the lithium oxide breeder module. The selected structure material is ferritic steel, with austenitic steel or titanium alloy Ti-4381 as a second choice. A flow of helium purge gas through a network of small channels in the breeder is used to remove the tritium.

The approach to the mechanical design is similar to that of the pressurized tube concept previously described for the reference helium cooled blanket. The coolant tubes are U-shaped with their end fittings at the back face of the module such that no welds for the coolant system are in the region of high neutron flux. The water coolant enters an inlet breeder system from the rear of the blanket and is distributed to each of the coolant tubes which run up the end of the module, turn 90° and then extend across the length of the module to a similar outlet header system at the other side. The module ends, sides and front face are made from panels actively cooled by water.

The coolant tubes are double walled; the inner and outer walls are sized to withstand full system pressure and 0.3 of the system pressure, respectively.

Figure 3.2-7. A schematic for the water-cooled blanket design.

Hoop load due to pressure in the inner tube is shared with the outer cylinder through ribs or fins, machined on the inner surface of the outer tube, which carry the heat from the breeder material to the coolant. The primary purpose of the double walled tube is to provide a double barrier between the high pressure water coolant and the lithium oxide breeder. The annulus between cylinders is connected to a sealed helium purge system to remove tritium bred in the lithium. The purge gas is monitored (on a sector basis) to detect the presence of water leaking into the annulus through the inner tube. This will permit the reactor to be shut down for sector replacement before pressure in the annulus can rise sufficiently to cause a crack in the outer tube and allow coolant to leak into the breeder.

The module outer wall is designed to withstand both the ~ 1 atm module internal pressure from the helium purge gas and the pressure from the coolant in the channels of the actively cooled wall. If higher pressures occur in the module a pressure relief unit in the module wall will be activated.

The coolant system is designed for 15.2 MPa (2200 psig). Blanket coolant inlet and outlet temperatures are 180° and 320°C, respectively. Gross system efficiency is estimated to be $\sim 34\%$, assuming no intermediate heat exchanger. The water cooled first wall panel design may be used for both the reference design and the alternate water cooled design; the inlet and outlet lines from the panels are connected to the blanket manifolds.

3.2.5.3 Backup

Should the reference helium cooled blanket and the alternate water-cooled blanket for STARFIRE both prove to have irresolvable design problems, a backup design which utilizes liquid lithium as coolant and breeder has been identified. This design would be based on the lithium blanket concept developed by Argonne during a joint ANL/MDAC study of blanket/shield designs for commercial tokamaks.⁽⁶⁾

A cross section of a typical module for the backup design is shown in Fig. 3.2-8. The blanket uses liquid lithium for both tritium breeding and cooling. The structural material would be a vanadium-base alloy, with ferritic steel as an alternate choice. The outer blanket consists of ~ 480 individual modules assembled in slab sections, with 4 to 6 modules per slab.

Figure 3.2-8. Backup: Liquid lithium cooled blanket design.

The inner blanket, basically a high-temperature shield, would also be lithium cooled with only modest pumping power penalty resulting from MHD losses. The front face of the blanket module will be used as the first wall of the plasma chamber.

The elongated, full flow cell concept shown in Fig. 3.2-7 features an ellipsoidal front face, to minimize the effective void space near the front surface and thereby reduce the thickness requirement. The module is fabricated from vanadium alloy sheet (~ 5 mm) with internal structural support. The inlet lithium coolant is manifolded to the front surface and through a thin flow channel adjacent to the first wall provided by a formed baffle. After the lithium flows over the first wall, it slowly circulates toward the back side of the module to the outlet manifold. The design pressure of the lithium coolant in the module is calculated to be ~ 2 MPa. The structural fraction is 7%; an effective breeding ratio of > 1.3 is attainable with a 40-cm-thick lithium module backed up by a 20-cm reflector. The headers that supply the modules are nested in the reflector region. Tritium levels are expected to be held to less than 0.1 kg for the entire blanket system. The inner blanket, which is lithium-cooled, provides an effective breeding ratio of ~ 0.6 from this region with $\sim 10\%$ coolant and 90% structure.

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3.3.2 Primary Loops

With the water-cooled first wall and inboard blanket, approximately 40 per cent of the thermal energy is in the form of 300°C heat. As described in Sec. 3.7, this energy is utilized in the power conversion system for feedwater heating and for steam generation. The remaining 60 per cent of the thermal energy is transported via the helium blanket coolant to the steam generators. The primary loops consist therefore of both helium and water circuits.

3.3.2.1 Primary Helium Loop

The primary helium loop(s) consists of the primary loop piping, steam generators (including reheaters), and helium circulators, as well as associated valving and instrumentation. Because STARFIRE operates steady-state rather than in a pulsed mode, a thermal energy storage system is not required as part of the primary loop. While detailed tradeoff studies regarding primary loop helium conditions have not been completed yet, preliminary calculations have been made for a helium pressure of 50 atmospheres, a helium ΔT of 200°C and a maximum helium temperature of 500°C. A higher operating pressure (and thus density) would yield lower pressure drops and, therefore, reduced pumping power requirements. However, this must be weighed against the problems associated with thicker structural walls in the various components as well as the safety implications of the higher pressure coolant.

As stated above, the helium loop interfaces directly with the steam loop in the steam generator with no intermediate loop between. Possible reasons considered for including such an intermediate loop were:

1. For thermal storage in a pulsed system.
2. To reduce the possibility of overpressurizing the primary loop in the case of a steam tube failure.
3. To reduce the permeation of tritium to the water/steam circuit.

Reasons for not incorporating an intermediate loop include design simplification, reduced cost, and increased performance.

Since STARFIRE is not a pulsed tokamak, the need for utilizing an intermediate loop for thermal load leveling has been eliminated. Regarding the consequences of steam ingress into the helium loop in case of a tube failure, a precedent for addressing this issue without an intermediate loop exists in the high-temperature gas-cooled reactor (HTGR). In that device, the worst of the several primary loop pressurization events considered is that of an offset shear rupture of a steam generator tube where the fluid in the tube is at a pressure of 2500 psi with the helium loop pressure at 735 psi. Leakage of water or steam from both ends of the failed tube is conservatively assumed to enter the primary coolant system as steam. Flow restrictions designed into the steam generator piping serve to control the rate at which steam (or water) can leak from the ruptured tube.

Three safeguards against steam inleakage have been provided in the HTGR design:

1. moisture monitors,
2. redundant pressure sensors,
3. steam generator relief system.

The moisture monitors detect moisture in the primary coolant and effect a reactor scram, steam generator isolation, and steam generator dump when a certain moisture level is exceeded. Pressure sensors cause the same set of actions to be taken if primary coolant pressure exceeds a certain set point. The steam generator dump system provides for expelling all of the contents of a particular steam generator to storage.

With these systems, the pressure rise in the primary coolant is limited to a value such that the pressure relief valves are not opened. Hence no primary coolant flows into the secondary containment.

In the case of the HTGR, the worst case considered results in a primary coolant pressure rise of 75 psi. For STARFIRE, a system can be designed with a manageable pressure rise since the primary coolant system volume is approximately one-half that of an HTGR of the same power rating and the steam pressure will probably be somewhat lower. With the primary loop pressure relief valves set to open at a slightly higher pressure, this accident would not be expected to release primary coolant into the reactor building.

While some modifications to the design details might be required, preliminary indications are that the basic approach to this problem as developed for the HTGR is valid for STARFIRE and that the primary loop pressurization accident is not in itself sufficient reason for incorporating an intermediate loop.

A calculation of permeation losses from the helium coolant through the steam generator was performed. The steam generator which is constructed of Croloy (Fe-2½Cr-1Mo) has a tube heat transfer area of 5440 m² (58,500 ft²), a tube thickness of 2.0 mm (0.080 in), and an average tube temperature of about 350°C. Although the He coolant does not contact the breeder directly, as a worst case it is conservatively assumed that a leak develops and the T₂O pressure is 10⁻² Pa. Assuming equilibrium with 1 Pa O₂, the T₂ pressure is then 10⁻¹³ Pa. Given the tritium permeability of 0.4 mm³(STP)·mm/s·m²√Pa (0.10 Ci/da·m√Pa), the tritium permeation loss is 0.10 Ci/day. It is noted that no credit was taken for a permeation barrier, which probably will reduce the rate by a factor of about 150. It is concluded that, due to the fact that the tritium is predominantly in the oxidized form, and due to the fairly low temperature, tritium permeation losses will be very small and an intermediate loop is not required.

Helium Loop Piping

One of the most important considerations for the primary helium loop piping is that of pressure drop and the associated pumping power. These can be minimized of course by increasing the pipe size and the number of parallel primary loops, however, both these approaches result in higher costs.

The effect of helium loop piping diameter and the number of loops is shown in Fig. 3.3-2. For this evaluation, a total piping length around the circuit of 125 m was assumed per loop. Since detailed piping layouts have not been made at this stage, it was further assumed that additional losses in each loop would be those associated with two sudden expansions, two sudden contractions and nine right angle turns. The flow rate was based on the assumption that 60% of the 3800 MW reactor power is derived from the helium cooled blanket with a 200°C blanket coolant ΔT. The strong effect of pipe diameter is shown in the figure. In order to keep the pumping power associated with piping losses to 1% to 2% of the total thermal power with a reasonable number (3 to 6) of parallel loops between the reactor and the steam generators,

Figure 3.3-2. Effect of pipe size and number of primary helium loops on pumping power for piping system only (steam generator and blanket pumping power not included).

pipe sizes in the 1.0 m to 1.75 m range will be required. While piping of this size is indeed feasible, the larger sizes probably represent the upper range of practical dimensions. As the pipe diameter approaches 2 m, considerations such as seismic design and thermal stresses during startup and shutdown transients may well become limiting. Also shown on this figure are the 100, 150, and 200 m/s velocity curves. Helium velocities in the Peach Bottom HTGR hot pipes approaches 125 m/s with no apparent problems, and there appears to be no reason to expect problems at velocities up to 150 m/s. The design will likely be in the 100 m/s range in order to keep the pumping power requirements within acceptable limits.

Steam Generators

In the steam generator, the heat from the reactor coolant is transferred to the power conversion system water/steam loop. Helical coil steam generators like those being developed for the high-temperature gas-cooled reactor (HTGR) will be utilized.

Preliminary sizing of steam generators for STARFIRE was performed in order to establish approximate space requirements. Units of 500 MWth each were assumed for these scoping calculations; when, during the next phase of the study, the number of primary helium loops are established, appropriate modifications will be made. Major steam generator parameters are given in Table 3.3-1. The overall dimensions of each steam generator are 4.7 m diameter by 9.2 m height. The modest pressure drop through the steam generator results in a pumping power of only about 1/2% of the thermal power.

Major considerations in selecting the number of steam generators and the number of primary loops are steam generator cost and size (impacts transportability), building size, piping layout and circulator size. These aspects will be addressed in the next phase of the study.

Table 3.3-1. Preliminary STARFIRE Steam Generator Parameters

Heat duty, MWth	500	
Shell side (helium)		
Inlet temperature, °C (°F)	500	(932)
Outlet temperature, °C (°F)	300	(572)
Pressure drop, MPa (psi)	0.028	(4)
Tube side (water/steam)		
Inlet temperature, °C (°F)	204	(400)
Outlet temperature, °C (°F)	427	(800)
Number of tubes	375	
Tube diameter cm (in.)	2.54	(1.0)
Overall diameter m (ft)	4.7	(15.5)
Overall height m (ft)	9.2	(30)

Helium Circulators

Scoping calculations were performed to establish the helium circulator characteristics. As this analysis was performed before other system parameters were firmly established, the following assumptions were made:

Reactor thermal power	3300 MW
Fraction of thermal power transported by helium circuits	0.60
Helium ΔT	200°C
Helium mass flow rate	~ 1900 kg/s
Circulator inlet temperature	300°C
Range of ΔP s considered	0.14 to 0.34 MPa (20 to 50 psi)

While these conditions are slightly different than the parameters presently established for STARFIRE, the general conclusions as well as the approximate circulator sizes are still valid. The basic conclusion is that feasible designs can be obtained with centrifugal compressors and, if the required pressure rise is small enough, with axial machines also.

The principal parameters for the main helium circulators are given in Table 3.3-2. Each circulator would pump one-sixth the total flow or 317 kg/s. Over the range of ΔP s considered, several different compressor configurations are possible. A single stage centrifugal compressor could cover the entire range. This unit could be coupled directly to an electric motor drive as the maximum speed required would be 3600 rpm. The driver would be a 3-phase synchronous motor with power supplied from a variable frequency solid state inverter. The motor would run on its own bearing system and be coupled to the compressor.

The compressor could also be a high speed axial flow unit if the required ΔP is low enough. With an axial flow compressor, multiple stages would be required at the higher pressure rises. A multistage compressor is undesirable because of potential blade vibration problems associated with variable speed operation.

Because the axial flow compressor requires a high speed driver, an axial steam turbine would be the probable choice in this power range. The turbine could be an integral part of the unit and run on the same shaft as

Table 3.3-2. Helium Circulator Parameters

Flow = 317 kg/sec
 T = 300°C
 P = 5.07 MPa (735 psi)

Type of Compressor	ΔP MPa (psi)	N rpm	N_s Specific Speed	Wheel Diameter m	Tip Speed m/sec	Power MW
Centrifugal	0.14 (20)	2096	100	1.94	213	13
Centrifugal	0.34 (50)	3600	87.57	1.77	334	32
Axial	0.14 (20)	6289	300	1.19	391	13
Axial	0.34 (50)	— Too high tip speed for single stage —				32

Centrifugal compressor tip speeds:
 305 m/sec — commercial maximum
 487 m/sec — experimental

Axial compressor tip speeds:
 243 m/sec — commercial
 305 - 457 m/sec — jet engine compressors

the compressor or the turbine could be an external driver with its own bearing system. For either a centrifugal or axial flow compressor the range of plant operating conditions would be accomplished with variable speed.

The bearing and seal system of the compressor would use either water or oil to provide a boundary between the high pressure helium and ambient. Isolation at the helium end would be provided by a buffer helium system utilizing a double labyrinth and clearance seals.

3.3.2.2 Primary Water Loops

Pressurized water which is used to cool the first wall and the inboard blanket exits the reactor at $\sim 300^\circ\text{C}$. A portion of this heat is utilized for feedwater heating in the power conversion cycle while the remainder

may be used to generate saturated steam. The details of this approach are discussed in Sec. 3.7. The pipe size and pumping requirements for the water loops are quite modest. While the number of parallel loops has not yet been established, a single water loop with a maximum water velocity of 10 m/s would result in piping about 0.7 m in inner diameter, with a pumping power of a few tenths of a per cent of the thermal power. Multiple loops would, of course, reduce the pipe size.

3.3.3 Auxiliary Cooling System

As discussed here, the auxiliary cooling system is concerned primarily with the blanket and not the first wall. The reason for this is that based on a preliminary assessment, it appears that a separately cooled first wall, which has lost its active cooling capability, could, with a quite modest ΔT , radiate its decay heat either to the blanket or to other first wall sections with functioning cooling systems. In this assessment, it was assumed that the plasma can be shut off rapidly enough such that excessive heatup (say no more than 100 to 200°C) of the first wall does not occur in the interval between loss of cooling capability and plasma shutdown. This is of course a function of the first wall geometry and whether or not the water coolant is present in the coolant passage. For a first wall thickness on the order of 0.1 cm, heatup rates with no cooling of about 25°C/sec could be experienced prior to plasma shutdown. This indicates plasma shutdown times of 4 to 8 seconds would be required, with longer times allowable for thicker walls. Because of the higher heating rates experienced by the limiters, a more rapid plasma shutdown may be required to prevent overheating in case of loss of limiter cooling. This will be assessed.

With a decay heat rate of 1.5 W/cc in the first wall, and an assumed thickness of 0.5 cm, the first wall can be maintained below 600°C for all reasonable postulated conditions through passive radiative cooling. Radiating from only one side of the first wall to the blanket (with assumed emissivities of 0.7), first wall temperatures of about 500, 550, and 600°C are obtained for blanket surface temperatures of 300, 400, and 500°C respectively. For radiation from both sides of the first wall, the temperatures could be reduced 50 to 75°C for the same assumed surface emissivities and sink temperatures. While the need for active first wall auxiliary cooling will be assessed further as the design progresses, the preliminary conclusion is that such a system is not required.

Regarding auxiliary cooling of the blanket, an evaluation of the requirements has not been made as yet, however this will be addressed during the next phase of the study. If it is required, additional complexity may be introduced into the blanket design, and this complexity will be a function of the particular conditions for which auxiliary cooling must be provided. In the following discussion, it is assumed that auxiliary cooling capability will be required for the blanket, and some of the auxiliary cooling system considerations are addressed.

The auxiliary cooling system provides an independent means of cooling the shutdown reactor and removing the decay heat produced by the blanket. The system will be used (1) when the main cooling loops are not available; or (2) when it is desired not to perform decay-heat removal with the main cooling loops. Examples of these conditions would be in case of a circulator failure, a helium pipe rupture, any other forced shutdown of a primary loop, or possibly during transport of a blanket segment. The auxiliary cooling system must provide adequate cooling and prevent the temperatures of the structure and breeder material from exceeding prescribed limits so that safe cooldown is ensured after any credible system failure.

While no sizing calculations have been performed for the auxiliary cooling system, it is anticipated that each auxiliary loop will incorporate one or more heat exchangers and circulators. The auxiliary loops will be designed to provide cooling in either a pressurized or a depressurized condition. In addition, the auxiliary cooling system must also be capable of providing safe cooldown of the reactor with a mixture of gases in the reactor coolant system, such as could result from accidents involving steam or air leakage.

Because it would be expensive and impractical to design the auxiliary cooling system to handle significant fractions of the normal heat load, it must be possible to shut down the reactor (turn off the plasma) on time scales on the order of seconds in case of a rapid loss of primary cooling capability. This shutdown would be initiated by the plant protection system logic which would monitor for the following conditions: high coolant temperature, high or low coolant pressure, high coolant moisture, low steam pressure, high containment pressure, loss of electrical power, or manual trip. It is expected that plant protection systems will correctly and rapidly sense anomalies in the blanket coolant system, shutting down the

plasma subsequent to loss of flow or loss of coolant. The primary post accident concern would then become decay heat removal in the blanket assembly. Afterheat generation rates are probably such that, if the plasma could be shut down rapidly, actuation of the auxiliary cooling system could be delayed for several minutes before structural damage would occur.

3.4 Toroidal Field Coil System

An important element of STARFIRE is the 12 coil superconducting toroidal field coil system, which is required to generate 5.6 Tesla on the 7.0 m plasma axis, with a peak-to-peak field ripple at the plasma edge of 1% (maximum). Spatial requirements for plasma, blanket, shield, and coolant manifolds dictate a clear bore about 15 m high and 10 m wide, resulting in a total stored energy of ~ 67 GJ.

3.4.1 Key Design Issues

Among the key issues to be addressed in the design are the following:

1. Selection of superconductor material.
2. Support of the coils against the out-of-plane (overturning) loads generated by the equilibrium field (EF) coils, located outside of the TF-coil envelope.
3. Cryogenic accommodation of the heat loads resulting from nuclear radiation and plasma disruptions and time varying poloidal fields.
4. Ease and location of manufacture. The feasibility of transporting completed TF-coils from a central manufacturing plant to a distant reactor site effects the need for spares and plant construction time. The elapsed time required to wind a single TF-coil, whether on-site or at a manufacturing plant will effect magnet cost.
5. Maximum operation reliability. This is achieved through a combination of sound, straightforward design, good manufacturing practice, and comprehensive component and system testing. An important design consideration is selection of a maximum allowable total fast neutron fluence to the conductor and insulation materials.
6. Maintainability. At best, the removal and replacement of a TF-coil following an in-service failure would be a costly, time consuming operation. The principal difficulties here would be removal of machine components surrounding the coil, and timely manufacture or shipping of a replacement. Nevertheless, the coils should be designed for relative ease of removal from the toroidal array.

7. Minimization of long-lived radioactive inventory and reducing the biological dose in the reactor building exterior to the TF coils to an acceptable level within a short time after shutdown.
8. The time and frequency of magnet annealing can have a major impact on reactor downtime.

3.4.2 Selection of Basic Design Features

Basic design features include the conductor material, current, geometry and support; coil cooling method; coil design; cryostat materials and design; and coil support.

3.4.2.1 Conductor Material

Various design studies have indicated that 10 Tesla is about the upper practical limit for employment of NbTi cooled with 4.2 K, one atmosphere liquid helium. Therefore, in the high field region (9-11 T) of an 11 Tesla STARFIRE TF-coil, selection of Nb₃Sn over NbTi appears to be warranted, based upon anticipated fabrication and metallurgical advances during the next 20 years or so. Nevertheless, Nb₃Sn is, and will remain an inherently brittle material, which can tolerate very little tensile strain ($\sim 0.1\%$). In order to minimize this difficulty, two expedients should be employed: (1) the conductor should be cabled and wound into the coil under zero or minimal pretension; and (2) the Nb should not be reacted with the Sn of the Cu/Sn matrix until after the cabling process has been completed.

As a possible stabilizer for the conductor, high purity aluminum has the advantage over copper of lower long-lived radioactivity. In addition, it has lower electrical resistivity than OFHC copper at high magnetic field, and thus offers superior stabilization performance for the superconducting magnets needed for advanced fusion reactors. For the environment of the TF coils, characterized by high fields, cyclic stress, and neutron irradiation, the electrical resistivity of pure aluminum is several times smaller than that of copper. In addition, the correspondingly higher thermal conductivity of aluminum adds to its performance as a superior stabilizer.

The poor mechanical properties of pure aluminum, with a yield strength an order of magnitude smaller than that of copper, presents a significant challenge to the mechanical design of reactor magnets. Co-drawing pure

aluminum with Nb₃Sn or NbTi or cabling strands of pure aluminum with strands containing Nb₃Sn or NbTi presents problems which have not been overcome.

If, however, steady-state operation of the tokamak reactor permits the use of a monolithic conductor rather than a cable, another concept may be employed. The success of the CELLO solenoid,⁽¹⁾ wherein a pure aluminum stabilizer was soft-soldered to a conventional Cu-NbTi superconductor demonstrates an early application of a technology that could readily be extrapolated to Nb₃Sn superconductor, now available in large quantities made with a copper matrix. For a fusion reactor, a Cu-NbTi or Cu-Nb₃Sn cable with copper to superconductor ratio of ~ 2 could be fabricated (and then reacted, in the case of Nb₃Sn). The cable could then be bonded to the monolithic pure aluminum stabilizer in the manner of the CELLO magnet.

Although copper has been selected as the stabilizer material in the preliminary design analysis for STARFIRE, aluminum is being examined as a serious candidate.

3.4.2.2 Conductor Current

Twenty-four kA has been selected based upon considerations of fabrication practicality, current lead heat load, and dump voltage (the latter being held to about 1 kV during a rundown with a ten minute time constant).

3.4.2.3 Conductor Configuration

A three-level cabled conductor design has been chosen for the following reasons:

1. To minimize coil heat loads resulting from plasma disruptions and time varying poloidal fields.
2. To provide a high cooled surface-to-area ratio, for cryostability.
3. As a practical, proven design for the fabrication of high current filamentary composite superconductor.
4. To minimize hoop tension experienced by the inherently brittle Nb₃Sn superconductor.

The conductor of each pancake (layer) is divided into four grades, the amount of superconductor in each being based upon the peak field experienced (5, 7, 9 and 11 Tesla); Nb₃Sn is employed only in the high field

(9-11 T) grade. The same number of strands, subcables and cables are retained throughout the layer, the strand size varying from grade-to-grade depending upon superconductor and substrate requirements. All conductor splices are, of course, made in the outer, curved portion of the coil.

For the reference design, cryostability of the conductor has been based simply upon the "quasi-static" criterion that, in a resistive region, the "effective" heat transfer rate to the coolant is 0.40 W/cm^2 . This criterion is applied to the 250 amp subcable, where 25% of the radially projected surface is assumed to be effective for heat transfer.

The 7-9 Tesla and low field (5 T) conductor configurations are depicted in the accompanying Figs. 3.4-1 and Fig. 3.4-2, respectively. The Nb_3Sn conductor for the high field (9-11 T) region has not yet been sized.

3.4.2.4 Conductor Hoop and Bearing Load Support

Inherent in a cabled conductor design is its limited ability to support hoop and transverse bearing loads (the latter occurring in the centerpost region of a toroidal field coil). In the selected design, the conductor is sandwiched between two stainless steel support strips which provide hoop load support. The conductor is flanked by two bearing load support strips, also sandwiched between the hoop load support strips.

Placing full width hoop load support strips both above and below the conductor and flanking bearing load strips helps to distribute the radial bearing stress experienced by the interturn insulation in the centerpost region. The conductor/support strip modules in the 7-9 T and low field regions are depicted in the accompanying Figs. 3.4-3 and 3.4-4.

The choice of stainless steel rather than an aluminum alloy for the support strips was made on the basis of its elastic properties. To limit the strain to a given level would require more than twice as much aluminum alloy as it would stainless steel. Consequently, the TF coils would be much thicker with the aluminum alloy support strips. Nonetheless, if the lower long-term radioactivity of the aluminum alloy relative to the stainless steel were an overriding consideration, certainly the conductor could be made to work with the aluminum-alloy support strips.

Figure 3.4-1. STARFIRE: TF-Coils. (24 kA conductor; 7-9 Tesla region.)

Figure 3.4-2. STARFIRE: TF-Coils. [24 kA conductor; low field region (0-5 Tesla).]

AVG. CURRENT DENSITY IN
MODULE 818 A/CM²

ALL DIMS. IN CM.

Figure 3.4-3. STARFIRE: TF-Coils. (Conductor/support strip modules; 7-9 Tesla field region).

AVG. CURRENT DENSITY IN CONDUCTOR
SUPPORT STRIP MODULE = 1034 A/CM²

ALL DIMS. IN CM.

Figure 3.4-4. STARFIRE: TF-Coils. [Conductor/support strip modules; low field region (0-5 Tesla).]

The thermal expansion coefficients of copper stabilizer and stainless steel support strips or of pure aluminum stabilizer and aluminum-alloy support strips provide a better match than other combinations of stabilizer and support strips. As a result, those combinations should experience less thermal strain during operation. However, this consideration is judged to be less important than those of Young's Modulus and long-term radioactivity.

3.4.2.5 Coil Cooling Method

Bath cooling has been selected in lieu of forced flow, based upon considerations of design simplicity and operational reliability. Forced flow cooling introduces complexities of conductor design, pumping, and parallel path manifolding which more than offset any potential benefits of improved heat transfer. In principle, forced flow cooling through simple well defined channels is well understood and is characterized by relatively high heat transfer rates (example: "hollow" monolithic conductor). However, the flow path geometries of cabled or braided conductor presently under consideration for large TF-coils are torturous. Proper understanding of such geometries will require considerable analytical and experimental effort. Certainly a far greater body of practical operational experience exists for bath cooling of large superconducting coils.

The possibilities of coolant path blockage or pump failure are reliability concerns for a forced flow system. Bath cooling on the other hand is inherently reliable so long as the coils remain fully immersed in liquid.

3.4.2.6 Coil Design

Conductor configuration, cooling method, cryostat structure and coil design are of course intimately interrelated design considerations. The essential features of the selected coil design are:

1. Helical (pancake) winding.
2. Interlayer coolant channels for bath cooling; cabled conductor provides diagonal channels for face, and conductor internal, cooling.
3. Variable height layers (pancakes) provide maximum use of center-post annular space.

4. Layers wound onto bobbin formed by the inner radius wall, and radial central web of coil helium vessel.
5. In the outer curved portion, the coil is self-supporting against hoop loads by an interturn stainless steel support strip. In the straight centerpost region, radial bearing loads are transmitted through stainless steel strips flanking each conductor turn.
6. All coil crossovers and splices are made at convenient locations in the outer, curved coil region.

3.4.2.7 Cryostat Materials

Structural aluminum alloy (6061-T6) has been tentatively selected for both the helium vessels and vacuum tanks of the toroidal field coils. The primary motivation is to minimize the biological dose that will be obtained in the reactor building external to the TF coils a short time after shutdown. A secondary advantage is weight reduction of the coil/helium vessel assembly. A tertiary, but possibly important advantage is that both the helium vessel and vacuum tank act as conductive shields for the TF-coils against plasma disruptions and time-varying poloidal fields.

The two disadvantages of aluminum as a structural material for this application are its low modulus of elasticity and its relatively high coefficient of thermal expansion, compared to stainless steel. Though further investigation is required, these limitations do not appear to outweigh the advantages of aluminum.

3.4.2.8 Cryostat Design

The following are selected features of the cryostat design:

1. Helium Vessel

- Material: Aluminum 6061-T6.
- Coil wound on "T" section weldment formed by vessel inner (minimum radius) wall and central radial web.
- The cross section of the magnet in the centerpost region has wedge shaped sides (30° included angle) and circular arcs at the centerpost and common dewar. The cross section in the outer curved coil region is rectangular, a transition occurring just outboard of the centerpost region. Each TF coil has an independent helium vessel.

- Coil radial (centering) loads are borne by a cold centerpost support cylinder.
- Sidewalls in straight, centerpost region include brackets to support radial (centering) loads of outer coil layers.

2. Vacuum Tank

- Material: Aluminum 6061-T6.
- TF-coil/helium vessels share a common vacuum enclosure, in centerpost region. Outer, curved portion of each TF-coil/helium enclosed by an individual vacuum barrier.

3.4.2.9 Coil Support Structure

The structure required to support the coil/helium vessel units against out-of-plane (overturning) loads has not yet been designed. A tentative evaluation of the overturning loads to be borne is shown in the accompanying Fig. 3.4-5. This is based upon employment of external plasma equilibrium (EP) coils.

3.4.3 Summary of Design Parameters

The basic imposed, selected, and calculated design parameters are summarized in Tables 2.2-7 and 2.2-8.

Reference

1. H. Desportes, et al., "Performance of CELLO Thin-Wall Solenoid," ICEC, Madison, Wisconsin (1979).

Figure 3.4-5. STARFIRE TF coil out of plane load.

3.5 Tritium and Vacuum Systems Considerations

The design objectives for the tritium and vacuum systems for STARFIRE are listed in Table 3.5-1. One of the most significant features is that every effort is being made to achieve a minimum tritium inventory, particularly in the fuel cycle. Achievement of this objective will impose stringent requirements upon the plasma impurity control method. It is noted that if impurity control can be achieved under conditions where it is possible to achieve a high ($\sim 10\%$) fractional burnup, vacuum pumping requirements are relaxed.

3.5.1 Vacuum Pumping System

The primary vacuum boundary of the STARFIRE vacuum system is at the inner wall of the shield. A pair of 60 meter long toroidal limiters pass circumferentially around the outer edge of the plasma region, and deflect ions from the scrape-off zone into the adjacent slots in the first wall. A cross-sectional view of these slots is shown in Fig. 3.1-2. The slots are 60 meters long and 20 cm wide and penetrate the first wall and blanket. The location and configuration of the limiter is optimized to maximize the probability of a molecule entering the slot after striking the limiter. Each slot contains a step to reduce neutron streaming as shown in Fig. 3.1-2. In the molecular flow region this step has minimal effects on the conductance, the only consideration being the additional slot length required to provide the step. A large volume plenum exists between the outside of the first wall blanket and the inside of the shield. Twenty-four 1.7 meter diameter, 1.5 meter long vacuum ducts penetrate the shield and provide access for liquid helium cryogenic pumps.

The liquid helium cryogenic pumps will be of the compound variety in which hydrogen and its isotopes are pumped by cryocondensation on a liquid helium cooled panel and helium is cryosorbed on a 4.2°K molecular sieve surface. The cryosorption surface on such a pump can become saturated with helium and will require regeneration after approximately eight hours of operation. Regeneration will be accomplished by stopping the liquid helium flow and allowing the sorption surface to warm whereupon the helium is released and pumped away by another pumping system.

Table 3.5-1. Design Objectives: Tritium and Vacuum System

1. Minimize Tritium Inventory

Plasma impurity control with high fractional burnup ($> 5\%$).

Minimum T_2 storage - requires redundancy and reliability in fuel reprocessing system.

Burn with DT ratio > 1 to achieve 20-40% reduction in tritium inventory in fuel cycle.

Minimize time for fuel recycle.

2. Achieve adequate vacuum pumping with minimum neutron streaming and with minimum tritium holdup.

Cryocondensation pumping for DT.

Rapid (~ 1 hr) pump regeneration.

Separate He pumping: cryosorption or possibly diffusion pumps.

3. Minimize Releases of Tritium to the Environment

Minimize tritium inventory in the coolant.

Utilize three levels of containment, include additional containment where appropriate, and provide for timely decontamination of potential tritium spills within containment.

Utilize permeation barriers.

Recover tritium from water.

Package and dispose of tritiated waste.

4. Minimize Potential for Exposure

Minimize tritium inventory in the coolant and in reactor building.

Utilize three levels of containment.

Minimize amounts of HTO and T_2O .

Utilize permeation barriers.

Design Emergency Air Decontamination System (EDS) to achieve R.C.G. levels in 48 hours.

5. Minimize Solid Waste Disposal

Reprocess tritiated waste.

Minimize contamination of organics.

Vacuum and transfer pumps, valves, etc. will have all metal seals.

If the impurity control system is to remove impurities at a rate sufficient to maintain a steady-state continuous burn plasma, 25% of the molecules which enter the slots at the first wall must be pumped. An analysis of this vacuum system pumping probability was performed. The slots were treated as rectangular ducts and were assumed to be in series with the circular vacuum ducts at the inner wall of the shield. The number and size of circular vacuum duct was assumed to be fixed for this analysis. A large volume plenum exists between the exit, the slots and the entrance to the circular ducts. This volume is assumed to have minimal effect on the system pumping probability. The final element in the vacuum pumping analysis is the cryogenic vacuum pump. A capture probability must be assigned to complete the calculations. A factor was also included to account for the directional nature of the molecules entering the slots, which enhances the system pumping probability.

The results of these calculations are presented in Fig. 3.5-1. The total system pumping probability is shown as a function of slot width for various cryogenic pump capture probabilities. It is seen that an overall pumping probability of 0.25 can be achieved for a 20 cm wide slot if a cryogenic pump capture probability of 0.5 is assumed. Such capture probabilities are beyond the present state of the art, but may become feasible as cryogenic pumping technology is advanced. Enlarging the slot widths beyond 20 cm, for a fixed vacuum duct geometry, will not improve the system pumping probability. However, widened slots will significantly increase the probability of a molecule entering the slots after striking the limiter surface and may thus reduce the required pumping probability. The calculated helium pumping speed for this system is 1.25×10^6 l/s at the limiter.

3.5.2 Fuel Cycle

A fuel cycle scenario for STARFIRE (Fig. 3.5-2) has been developed consistent with the design objectives (Table 3.5-1). Starting with the plasma chamber, the first item encountered is the limiter-vacuum system. The DT is pumped by cryocondensation pumping and the He is pumped by cryosorption pumping or possibly by diffusion pumping. Recent results⁽¹⁾ indicate that an essentially clean separation between He and hydrogen isotopes can be achieved with a cryocondensation/cryosorption pump configuration. The helium is then passed to the Tritium Waste Treatment (TWT) subsystem for

Figure 3.5-1. Vacuum pumping requirements for limiter-vacuum system.

removal of trace amounts of tritium. The DT is regenerated from the condensation pump (two sets are operated in tandem), pressurized to about 0.1 MPa (1 atm), and transferred to the Fuel Cleanup Unit (FCU), where condensibles are removed to levels of < 1 wppm.⁽²⁾ The tritium in the breeder blanket will be recovered as T_2O from He coolant or from a He purge stream, if a solid breeder is used. Upon reduction, the bred tritium is passed to the FCU. The purified fuel is then isotopically enriched and prepared for re-fueling. The fuel processing subsystems are contained within gloveboxes, which in turn are located in a separate tritium facility building.

The tritium and vacuum parameters are listed in Table 3.5-2. The total tritium inventory is presently estimated to be approximately 2.5 kg.

References

1. J. L. Anderson, D. O. Coffin and C. R. Walthers, Los Alamos Scientific Laboratory, personal communication (1979).
2. J. L. Anderson and R. H. Sherman, "Tritium Systems Test Assembly Design for Major Device Fabrication Review," Los Alamos Scientific Laboratory report LA-6855-P, (June, 1977).

Table 3.5-2. Tritium Facility and Vacuum System Parameters for the STARFIRE

Torus Evacuation System Parameters

Fractional burnup	0.11
Evacuation Volume (m ³)	902.8
Gas Load-Limiter (Pa-m ³ /s)	9.89E 01
Total Pump Speed at Limiter (m ³ /s)	2.6E 03
Total Rated Pump Speed-DT (m ³ /s)	2.0E 04
Total Rated Pump Speed-He (m ³ /s)	1.2E 04
Cryopanel Surface-Lim. (m ²)	2.4E 02

Tritium and Fuel Processing Parameters

Tritium Burnup per Day (g)	495.40
Tritium Exhaust per Day (g)	3.99E 03
Tritium Fueling per Day (g)	4.48E 03
Tritium Input per Day (g)	4.48E 03
Helium Exhaust per Day (g)	659.
Tritium Bred per Day (g)	498.
Tritium Decay per Year (g)	143.
Doubling Inventory (g)	3.60E 03
Doubling Time (years)	5.
*Breeding Ratio	1.005
Tritium Inventories (g) *****	
Vacuum Pumps	330.
Storage	495.
Surge Tank	83.
Liquefaction Unit	45.
Cryogenic Distillation Cascade	105.
Fuel Preparation	240.
Breeding Blanket	1000.
Tritium Recovery System	250.
Total Tritium Inventory	2550.

Emergency Air Detritiation System (E.D.S.)

Maximum Conceivable Release (g)	818.10
Initial Conc. (μCi/M**3)	3.93E 07

Table 3.5-2 (Continued)

Emergency Air Detritiation System (E.D.S.) (cont'd)

*Cleanup Time (hours)	48.00
*Volume of Reactor Bldg. (m ³)	2.00E 05
*Decontamination Factor	1.00E 06
*Cleanup Efficiency	0.900
E.D.S. Flow Rate (m ³)	17.5
E.D.S. Flow Rate (% Bldg. vol/min)	0.52
*T Level Stacked (μCi/m ³)	50.000
Tritium Release (Ci)	10.000
Vent Rate (m ³)	3.33E 00
Tritium Vented to Env. (Ci)	1.50E 00

Tritium and Vacuum Systems Costs

Tritium Recovery (\$)	2.49E 06
Piping (\$)	1.82E 06
Isotopic Separation Unit	1.93E 06
Gloveboxes and Purifiers (\$)	6.10E 06
Misc. T Facility Costs	2.39E 06
Pellet Fueling (\$)	8.70E 06
Total Fuel Processing (\$)	2.10E 07
Cost - E.D.S. (\$)	1.95E 07
Cost-Limiter Vacuum System (\$)	1.7E 07
Total Tritium + Vacuum Systems (\$)	6.0E 07

3.6 Maintenance

3.6.1 Maintenance Approach

The ability to maintain the STARFIRE reactor rapidly with a minimum of radiation exposure to maintenance personnel is key to its acceptance by the utilities and public.

The remote maintenance approach was chosen for in-reactor operations because of the likelihood that current regulatory radiation limits will be significantly reduced and because the reactor hall will be exposed to tritium by permeation and releases during maintenance of the coolant, fueling, vacuum and tritium systems. Some activation may also result from neutron streaming. Use of remote equipment will permit maintenance with a minimal cool-down or clean-up period.

The design philosophy being followed is to minimize the radiation levels within the reactor building; to design all components for complete remote maintenance, and to identify contact maintenance operations where personnel can safely be used with significant economic savings.

All components within the reactor building are replaceable. Some are replaced on a scheduled maintenance basis while others are designed for the life of the plant and are replaced only in the event of failure. Items designed for the life of the plant include the overhead crane, TF coils, EF coils, coolant piping, reactor support structure and radiation shielding. The blanket assembly, impurity control components, rf launchers, pumps, valves, fueling mechanism, power supplies, etc. are replaced on a scheduled basis. Spares are provided for all components with expected high failure rates, so that as one part is removed a pretested replacement is available so reactor operation can commence while repairs to the damaged components are being made. The spares for the superconducting EF coils trapped below the TF coils are stored in place so reactor disassembly is unnecessary in event of a failure. These coils are designed for life of plant but the consequence of their failure suggests in place spares are prudent.

The number of different maintenance operations planned in the reactor building are minimized by using a component "remove and replace" approach. This permits each maintenance action to be preplanned and designed for use with simple push, pull, etc., operations. This approach increases the

confidence in the speed of maintenance operations and simplifies maintenance equipment design requirements. Once the damaged or end-of-life components are removed from the reactor they are transported to a hot cell where more time is available for checkout, repair or disposal. The hot cell will have extensive maintenance capability for testing, component replacement, cutting, welding, machining, pinpoint leak location, and low-Z coating repair.

Redundancy is planned for reactor auxiliary subsystems to permit continued operation of the plant until a scheduled maintenance period or until the component can be replaced in-service. The particular components where redundancy is planned will be defined as the design progresses; however, current plans include redundant power supplies, vacuum pumps, rf launchers, some valves, pumps and fueling mechanisms.

Availability goals have been established as 85% for the reactor and 75% for the complete plant including the reactor. Allocations of permissible time-to-repair and time-between-failures have been made for major subsystems to serve as a basis for design of the components and maintenance equipment. The maintenance scenario incorporates the current utility practice of shutting down annually for one month of maintenance and a 4 to 5 month shutdown every 5 to 10 years for turbine repair. The resultant permissible downtime goal per calendar year has been allocated as 30 days for scheduled maintenance of the entire plant, 21 days for unscheduled maintenance of the balance-of-plant and 40 days for unscheduled maintenance of the reactor, auxiliary subsystems and maintenance equipment. Convenient preventative maintenance and repair of redundant components is included as part of the maintenance scenario during these unscheduled outages.

3.6.2 Maintenance Features

The design is being developed to keep the top and sides of the reactor clear for access by maintenance equipment. Components are also being combined where practical to minimize the number of assemblies and improve access. Examples of combined components include the TF coil room-temperature dewar and shield which provides the EF and TF magnet structural support system, and the rf system which provides startup and plasma current drive functions.

The blanket is replaceable as a 1/24 sector of the torus. Each 1/24 sector incorporates an integral first wall, limiter, rf duct and air bearing pad that is removed with the blanket. The coolant is manifolded through separate loops for each 1/24th of the reactor to permit leak isolation to individual sectors. The helium coolant lines and the water coolant lines to the rf duct and limiter utilize mechanical disconnects. The life goal for limiter and rf duct is 4 years and the blanket life goal is 8 years. After four years the blanket sector is replaced. The used sector is refurbished in the hot cell by replacing the limiter, rf duct and low-Z coating. Periodic in-situ replacement of the low-Z coating may also be required.

Blanket module leak detection will be accomplished by sequentially reducing the pressure in individual blanket sectors and monitoring the change of the partial pressure of helium in the plasma chamber with a gas analyzer. This technique requires valves that isolate sectors from the primary coolant loop. Leak detection of the limiter first wall and rf launcher water system will also rely on reducing the system pressure of each sector sequentially and denoting changes in the detected leak rate. Leak detection of the shield system to the plasma chamber system will be accomplished by injecting helium into a cavity between redundant seals. The blanket leak detection system is dependent on the particular design that is developed and will be modified as necessary.

The shield is designed to last the life-of-plant and will be replaced only in event of unscheduled failures. A shield door is provided that permits access to the blanket. The door is sealed with redundant seals that are shielded locally to reduce the radiation damage. The seals are replaced each time the door is opened. The basic shield consists of sectors welded together to form the vacuum boundary. Cutting and rewelding the poloidal seals is required if shield sector replacement is required.

The TF coils are also life-of-plant components. They utilize a common welded dewar in the center post region. Coil replacement time is not significantly impacted by the welded dewar. Annealing of the TF coil is planned every 10 years to reduce the effects of radiation damage in the stabilizer.

The EF coils are also life-of-plant. EF coils inside the TF coil are copper and are segmented to permit removal. An elevation system is provided to raise and lower the outer coils for access to the blanket. The external

EF coils are superconducting. Those on the top and sides of the reactor can be removed for replacement. Spare EF coils are provided for those trapped below the TF coils. In event of failure, the coil is cut out and the spare raised into position.

The vacuum system utilizes redundancy to improve the probability of continuous operation between scheduled maintenance periods. Twenty-four cryosorption pumps with isolation valves are provided of which approximately 20 are required for operation. External valves and pumps leading to the tritium processing system utilize redundancy.

The power supplies are assumed fully redundant and include a replacement capability during operation. The power supply systems are located outside the primary confinement building and can be repaired during plant operation.

Failed redundant heat transport system components, with exception of a few in-line valves, can be replaced during reactor operation using remote/hands-on maintenance equipment.

3.6.3 Hall Maintenance Functions

A preliminary listing of scheduled maintenance functions is provided in Table 3.6-1. The number of scheduled actions at the reactor island have been held to a minimum. The scheduled maintenance functions of the auxiliary systems can, in part, be done during reactor operations.

The planned unscheduled maintenance functions include all component assemblies making up the reactor. The unscheduled items are shown in Table 3.6-2. Both Tables 3.6-1 and 3.6-2 will be updated as the design progresses and the subsystems are better defined.

3.6.4 Hot Cell Operations

The functions planned in the hot cell are listed in Table 3.6-3.

3.6.5 Maintenance Time Allocations

A preliminary listing of the scheduled maintenance actions is shown in Table 3.6-4. The assumed critical path is shown with the presumption that all other maintenance actions can be accomplished in parallel operations. This listing will be updated as the program progresses.

Table 3.6-1. Scheduled Maintenance Functions

Reactor Island

- Shield door seal replacement
- rf wave guide window replacement
- Blanket sector replacement
- Vacuum pump/isolation valve replacement
- Fueling system replacement
- TF coil annealing

Auxiliary Reactor Subsystems

- Coolant loops
 - valve replacement
 - pump replacement
 - Power systems
 - electrical switch replacement
 - electrical energy storage component replacement
 - power conversion system replacement
 - Maintenance equipment replacement
 - Tritium processing
-

Table 3.6-2. Unscheduled Maintenance Functions
(excluding scheduled maintenance functions)

- Power lead replacement
- Coolant line replacement
- Heat exchanger replacement
- EF coil replacement
- TF coil replacement
- Shield replacement/repair

Table 3.6-3. Hot Cell Functions

- **Component testing**
 - vacuum integrity
 - electrical function
 - configuration
 - leakage
- **Repair**
 - cutting
 - welding
 - machining
 - cleaning
- **Reactor retrofit**
 - blanket
 - limiter replacement
 - rf launcher replacement
 - low-Z coating replacement
 - module replacement
 - tritium processing system
 - vacuum system
 - coolant systems
 - cryogenic systems
 - power system
- **Maintenance equipment**
- **Decontamination/disposal**

Table 3.6-4. Scheduled Maintenance

Action	Frequency (yr)	Replaced/ Outage (%)	Time-to- Replace (days)	Critical Path
Reactor shutdown and startup	Annual	N/A	2	*
Shield door seal replacement	Annual	25	2	*
RF wave guide window replacement	4	25	6	*
Blanket replacement	Annual	25	20	*
Vacuum pump replacement	Annual	16	6	
Fueling system replacement	Annual	25	6	
TF coil anneal	10	100	120	(1)
Coolant loop component replacement	Annual	25	14	
Power system component replacement	Annual	50	14	
Maintenance equipment repair	Annual	N/A	N/A	

* These operations must be done sequentially, other operations are assumed to be performed during the critical path time frame.

(1) Compatible with 10-yr shutdown for 16 to 20 weeks for turbine repair.

The unscheduled maintenance item listing is shown in Table 3.6-5. The values shown for Mean-Time-To-Replace (MTTR) are assumed values based on past studies. The average downtime per year is then assigned to each subsystem based on the perceived relative reliability and complexity of each subsystem. The result is a permissible failure rate per calendar year. These values can then be used by subsystem designers to define the redundancy and design margin requirements for the subsystems. An accurate assessment of the actual failure rates can only be derived after subsystems are defined as to the number of components, redundancy provisions, the types of failure modes and failure rate per failure mode have been established. Our program will utilize historical data where possible to assess feasibility of meeting the requirements of Table 3.6-5.

Table 3.6-5. Unscheduled Maintenance (Forced Outage)

Subsystem	MTTR ^a	Average Downtime per Year (days)	Permissible Failure Rate per Year ^b	Availability (%)
Shield sector	30	2	0.04	5
RF system	30	3	0.1	7.5
Blanket system	30	15	0.5	37.5
Vacuum system	6	2	0.3	5
Fueling system	4	2	0.5	5
Magnet system - TF	350	2	0.006	5
Magnet system - EF	60	1	0.016	2.5
Primary coolant	4	2	0.03	5
Auxiliary coolant ~ H ₂ O	4	2	0.5	5
Auxiliary coolant - GHe	4	2	0.5	5
Auxiliary coolant - LN ₂	4	2	0.5	5
Power supplies	4	2	0.5	5
Maintenance equipment	2 ^c	<u>1</u>	0.5	<u>2.5</u>
Total:		40		100%

^aIncludes startup and shutdown.

^bFailure resulting in an outage.

^cOccurs during outage only.

3.6.6 Reactor Building

The reactor building is currently being developed. Major building requirements are that it 1) contain any release of tritium from the reactor, 2) withstand overpressure requirements, 3) protect plant personnel from radiation, and 4) provide a balance between capital cost (size) and space for efficient maintenance.

Tritium containment requires the building to be sealed with a welded metal liner and be able to withstand a 3 psi (0.2 atm) overpressure resulting from a credible accident condition where a portion of coolant and cryogenic

helium are spilled. Tritium containment also requires incorporation of airlocks in the building to permit transfer of maintenance items.

Biological shielding will be required in the reactor hall walls to protect plant personnel from radiation leakage. The primary effect of increased biological shielding thickness is in building capital cost.

The size of the building will affect both initial building costs and the reactor availability caused by the buildings enhancement or degradation of maintenance operations. Both a conventional round containment building and a rectangular building are being considered. The round building provides a more efficient structure for pressure containment but requires use of a polar crane which is somewhat more difficult to repair. The rectangular building offers advantages for crane coverage and possibly provides more efficient floor space utilization.

The building overpressure requirement is not excessive for either building shape and indicates the building layout should be developed using maintenance operational requirements and building cost as a criteria. At this time building costs have not been considered in detail, but it is expected that reactor availability effects of the building layout will override building cost differences. A building layout is being prepared for each concept to compare the maintenance flow. The initial rectangular building layout is presented here to show the major items being considered, it incorporates the repair cell facility for major reactor components into the same building as the reactor (see Fig. 3.6-1). This may simplify reactor disassembly/reassembly and component transport operations and, therefore, enhance plant availability. Activated blanket modules and all highly contaminated sources of radioactivity are confined to a single steel-lined reactor containment building that utilizes common services, maintenance equipment, cranes and the HVAC system. The concept is also motivated by other advantages including: use of more conventional rectangular structures and possible use of two cranes on the same rails, each of which can service the entire reactor building with no blind spots to either crane. The reactor is installed at the approximate center of the bay with ample space around the reactor provided to allow blanket modules to be remotely removed using a remote controlled transfer machine that is capable of lifting the blanket

Figure 3.6-1. STARFIRE

Figure 3.6-1. STARFIRE facility.

module, radially removing it from the vessel assembly and transferring the module to the repair cell.

A large area below the floor level in the reactor hall will house process equipment modules. These modules allow rapid removal and replacement of process subsystems. The servo manipulator system assists in disconnecting the electrical and utility disconnects and the overhead crane is used to lift the process module up and out of the floor and to place it upon a remotely controlled cart to be transferred into the repair cell for maintenance. A qualified spare module is then reinstalled to allow reactor start-up in the shortest possible time. All modules are fitted to installation gages to insure proper mate-up in the process cells under the floor.

The very nature of the remote manipulator requirements for the reactor will require a flexible system capable of reaching into the reactor along radial lines between the TF coils. The overhead shield structure prevents overhead manipulator bridge units from reaching horizontally into the doorways of the shield structure. Also, piping, coils and vacuum pumps installed above the reactor essentially dictate the need for a circular servo manipulator system that can run around the reactor on rails and reach into all sectors of the reactor between the TF coils.

Service galleries below grade and outside of the containment walls will allow distribution of utilities and services to the reactor from outside using special penetrations in the wall.

Above the repair cell is located a crane maintenance balcony that is able to receive the large 400 ton crane and to provide a shielded area for contact maintenance to the crane and other selected equipment. Large shield doors permit this area to be isolated from the reactor during reactor operations. Components as large as the poloidal coils may be transferred to this shielded area.

Internal shielding would be provided in the reactor building as necessary to allow repairs to be made on components in the repair cell while the reactor is running. The anisotropic nature of the neutron radiation in the reactor hall will require investigation for an effective shielding design.

As shown in Fig. 3.6-1, the major features of the rectangular reactor building concept include the following:

- Conventional overhead bridge crane(s) with minimal rail span.
- Reactor disassembly, repair and inspection integrated into a single facility.
- Common services, maintenance equipment and personnel utilized.
- Construction door allows removal of failed TF coil from the reactor hall.
- Blanket modules readily removed and transferred in repair cell.
- Containment structure provides shielding as well as contamination barrier (steel lined) for airborne activity that might result from normal operations as well as credible postulated events.
- Module replacement of all process equipment within the reactor hall is envisioned to simplify replacement operations as well as to minimize reactor downtime.
- Mobile force reflecting servo-manipulator systems will act as a substitute for man in the reactor hall environment and it is anticipated that by the year 2005 these units will be extremely compact and controlled by advanced computerized systems.
- Remote operations performed by machines will reduce personnel radiation exposure and will allow the plant to be operated by a small well-trained staff.
- Minor repairs can be performed in the reactor hall using remote systems without shutting down the reactor.

3.7 Facility Considerations

During fiscal 1979, major reactor system concepts are still undergoing design evolution and improvement. Consequently, a major facility consideration has been the identification of the balance of plant (BOP) functions and requirements. This was accomplished by identifying the interface between reactor systems and their BOP support systems, and their required functions. Major BOP support functions include steam and electric power generation, heating, cooling, fluids and gases, building services, building structures, radwaste handling and storage, maintenance support, and electrical power supply. Special attention has been placed toward minimizing potential tritium contamination and releases.

In establishing BOP design criteria, current design parameters were examined to determine their potential impacts in terms of BOP design considerations including building design, cooling requirements and steam cycle efficiency, etc. In addition, major earlier tokamak facility studies were reviewed to assess potential facility design concepts and approaches.

On the following pages, the BOP systems and buildings are described with emphasis placed on system/building definitions and design criteria. Trade-off studies and evaluation necessary to support the BOP conceptual design effort are also identified. These include sizes and configurations of all buildings and steam cycle efficiency.

The bulk of the BOP conceptual design effort is scheduled for the next fiscal year.

3.7.1 Review and Analysis of Previous Tokamak Facility Conceptual Design

As a starting point for the BOP, a number of previous conceptual design studies⁽¹⁻⁷⁾ were reviewed to determine their relevance to this conceptual design study. Some of the major similarities and dissimilarities are discussed below.

3.7.1.1 Site Planning and Building Arrangements

In all cases the reactor building is the focal point of the overall facility. This is surrounded by other elements such as the heat exchanger

building, reactor auxiliary building, electrical building, maintenance building with hot cells, and radwaste building.

Near, and generally contiguous to, the reactor building are the turbine building, control building, administration building, and emergency diesel generator building.

Around this building complex are located the electrical switchyard, cooling towers, road system, parking and security provisions, rail spurs and water reservoirs.

3.7.1.2 Power Conversion Cycle

The choice of a power conversion cycle for a fusion generation plant depends largely on the temperature capability of the blanket. Blanket materials are exposed to an intense neutronic environment and subjected to thermal induced stress, hence are temperature limited for the near future.

Most tokamak-type designs reviewed have limited steam power cycle temperature and pressure because of the blanket temperature limitations and heat exchange differences through intermediate loops to the steam generator. This has resulted in a range of turbine steam conditions of 565 psia to 2400 psia and temperatures to 950°F (510°C) in previous studies.

In addition to the steam cycles incorporated in most designs, UWMAK III also utilized a closed loop helium gas turbine cycle. The gas turbine inlet temperature is the most important parameter for this cycle and high efficiencies are obtained only with gas temperatures around 1600-1800°F. In the UWMAK III design, the temperature was constrained by a blanket lithium outlet temperature of 1800°F.

The current STARFIRE design utilizes a two loop coolant system. The blanket coolant is helium while the first wall coolant is heavy water (D₂O). The helium is limited to an outlet temperature of 932°F (500°C) and the pressurized heavy water outlet temperature is limited to 300°C. Both coolants are used as the heat source for the steam conversion cycle.

The helium supplies about 60% of the reactor thermal energy to the steam generator without an intermediate loop. As a first design approximation, the steam generator produces steam at 1800 psig and 800°F (427°C) to the high pressure turbine with reheated steam at 800°F.

The heavy water coolant provides feedwater heating through a heat exchanger system. There is an excess of first wall thermal energy available that cannot be efficiently used for feedwater heating, and this will be used to generate steam at the same pressure as the high pressure turbine exhaust. This steam will be combined with the HP turbine exhaust before it is reheated and goes to the low pressure turbine.

The final selection of blanket and first wall configuration and capabilities and their integration with the power conversion system requires several iterations to obtain a compromise between the blanket/first wall and the steam cycle configuration to obtain the optimum overall plant thermal efficiency.

3.7.1.3 Cooling Water System

Most previous fusion reactor studies were based on the use of wet cooling towers with the exception of UWMAC III which was based on dry cooling towers.

Based on the projection of continued growth in the use of water throughout the country, it appears that large amounts of water may not be available in most parts of the country 30 to 50 years from now. However, the STARFIRE cooling water system design will be based on the use of wet cooling towers to make the study more comparable to existing power generating plants and to prior fusion design studies. Using dry cooling towers would increase the plant cost, decrease cycle efficiency and affect other plant systems.

3.7.2 Plant Siting

The following site conditions have been assumed for the STARFIRE facility. These site conditions should cover a majority of potential power plant locations in the continental United States.

- Location: Middletown, U.S.A. (Kansas City, Mo., for cost estimate)
- Siesmic: Zone 2 of the Uniform Building Code
- Site Size: 1000 acres
- Ambient temperature: 102°F (39°C) for cooling
-10°F (-23°C) for heating
- Altitude: 2000 ft (610 meters) maximum
- Heat rejection system: wet cooling towers
- Water supply: adequate to support normal requirements

- Electrical utility transmission: 100-200 miles from site
- Transmission grid: 3-phase short circuit capacity 5000 MVA minimum
- Construction power: 35 kV or higher at distance of 5 miles maximum over 5 miles, on-site generation
- Operator training facilities: mockups and simulators will be located off-site
- Component assembly: If feasible, major components, such as coils, will be manufactured and assembled off-site and shipped to construction site.

3.7.3 Buildings

The STARFIRE facility building elements have been identified as follows:

- Reactor	- Electrical
- Hot Cell & Decontamination	- Cryogenics
- Tritium Reprocessing & Storage	- Control
- Heat Exchanger	- Radwaste
- Reactor Auxiliary	- Maintenance & Storage
- Turbine - Generator	- Administration
- Auxiliary Boiler	- Security
- Diesel Generator	

These building elements were identified from the evaluation of the BOP interface supporting requirements of the STARFIRE design. Each building element does not necessarily represent a separate physical building in the final BOP conceptual design. Rather, these building elements will be used as basic plant layout segments to evaluate the functional relationship of various building elements, to study various plant layout concepts, and to obtain a cost effective building complex arrangement.

Building design depends upon the functions that the building should provide to support systems and components which are housed in it. Factors such as seismic requirements, tornado winds, tornado generated missiles, floods, tritium confinement, overpressurization, and radiation shielding will also impact building design. In addition, there are the considerations of number and sizes of equipment, laydown space requirement, maintenance philosophy, and costs.

The reactor building is the focus of the building arrangement. The reactor building will house the tokamak fusion reactor and its main supporting systems such as vacuum pumping, primary coolant piping, rf wave guides and cryogenic system piping. The size of the reactor building is of major concern because of its potential cost implications.

The basic functions of the reactor building are as follows:

1. House the tokamak reactor with sufficient space for handling systems to allow for construction, operation, and maintenance.
2. Maintain structural integrity and pressure boundary for tritium confinement under normal and postulated accident conditions.
3. Provide necessary radiation shielding to protect plant personnel inhabiting adjacent building spaces during reactor operation.
4. Maintain the radiation dose exposure at the plant boundary to as low as reasonably achievable.
5. Provide space for those support systems that should be located in proximity to the tokamak reactor.
6. Accommodate necessary support functions and design features such as handling of large reactor components, the entry/exit services and remote maintenance provisions.

Due to intense neutron radiation generated by the fusion reaction and its associated activation problem, the size of the reactor building is critically dependent upon reactor maintenance philosophy as well as the plasma heating method. A study will be made in FY 1980 to closely look at means of minimizing the size and height of the reactor building.

Design of the hot cell depends largely upon the remote maintenance approach. The functions to be accomplished in the hot cell are to service the first wall and blanket modules and to provide temporary hot storage.

The tritium hazard potential and its siting implications are well recognized in the STARFIRE project. In addition to a multiple barrier confinement approach, it is a design goal to minimize the on-line tritium inventory to reduce tritium release potentials. The building design, as well as other BOP system design, including HVAC, will be utilized to enhance tritium confinement and cleanup features.

The radwaste building will contain the space, equipment, and systems for the collection, monitoring, classification, treatment, packaging, decontamination, and interim storage required for all radwastes generated within the facility. Various portions of the building may require hardening to protect against the accidental release of radioactive contaminants to the atmosphere.

The other buildings will be of more conventional design.

3.7.4 Balance of Plant Mechanical Systems

The BOP mechanical systems that will be required to generate electrical power, support the reactor and power generation systems, maintain temperature control within buildings, and provide general power plant support services are as follows:

- Steam and power conversion	- Plant service
- Plant water supply	- Radwaste
- Plant closed cooling water	- Helium storage
- EF coil cooling	- Nitrogen storage
- Heating, ventilating & air conditioning	- Plant service
	- Fire protection

The specific BOP mechanical system requirements such as flow rates, temperatures, pressures, temperature/pressure limits, purity etc. for the systems listed above are yet to be determined.

The steam and power conversion system consists of components of the more conventional design used in large central generating stations. Turbine plant equipment will be selected to provide operating economy with safety and reliability. The system will be designed to perform the following functions.

1. Produce approximately 1000 MW of net electrical power output to the transmission grid.
2. Provide continuous operation at 105 percent of the rated steam flow.
3. Provide regenerative steam from turbine extraction points to improve the thermal efficiency of the cycle.

4. Condense steam and return it as high quality, heated feedwater to the steam generators.

5. Provide steam to the station auxiliary steam system.

The plant water supply system will supply water to meet the operating needs of the facility during normal, abnormal and emergency modes. An emergency water supply will be maintained, and provisions made for replenishing the fire protection water storage. The system will also provide the source of demineralized water for the makeup requirements of the plant, and the source of potable water.

The system will furnish water to the following other systems:

- Steam and power conversion	- Plant service
- Plant closed cooling water	- Radwaste
- EF coil cooling	- HVAC
- Klystron cooling	- Fire protection

The plant closed cooling water system will provide a plant wide cooling system for heat rejection to the atmosphere. The system will be designed on the basis of a maximum dry bulk temperature of 102°F (39°C) and a maximum altitude of 2000 ft (610 meters). The system will be capable of providing heat rejection for plant normal and abnormal operating loads and for emergency shutdown conditions. Wet cooling towers will be used as the major heat rejection method. Another major design consideration will be to minimize any potential for tritium contamination of this system.

This system will interface with the following systems:

- Steam and power conversion	- Helium purification
- Plant water supply	- Tritium distribution, storage, and reprocessing
- EF coil cooling	- Emergency tritium cleanup
- RF heating	- Radwaste
- Vacuum pumping	- Plant HVAC
- Cryogenic	- Electrical power supply
- Limiter cooling	

The plant closed cooling water system will consist of wet cooling towers, pumps, heat exchangers, filters, demineralizers, surge tanks, valves, piping and instrumentation as required.

The radwaste system will provide for the collection, monitoring, classification, treatment, packaging, decontamination and interim storage of the liquid, gaseous, and solid radioactive wastes generated by the facility. The system will consist of several subsystems: liquid radwaste, gaseous radwaste, solid radwaste, and decontamination. The equipment will include radioactivity monitors, tanks, pumps, filters, canning equipment, etc. as required. This system is required to ensure that any radioactive releases to the environment will be as low as reasonably achievable and will not exceed the release levels as specified by applicable regulations. Low-level liquid wastes will be collected, monitored, treated, and stored until certified for release to the environment, or recycled if practical. High-level liquid wastes will be collected, treated, stored, concentrated and/or solidified. Gaseous wastes will be collected, monitored, treated for removal of tritium, and released. Solid wastes will be collected, monitored, decontaminated, compacted, packaged for transport, and stored until transported.

The pressurized helium storage system will receive, store, and maintain an inventory of helium to provide for the replenishment and makeup requirements. The system will also provide for the recovery, storage, and transfer of gaseous helium from the reactor primary cooling system and/or the cryogenic system during shutdown.

The HVAC system design will provide for heating, cooling, ventilation and temperature control for comfort and process requirements, control of airborne contamination, and environmental monitoring for radioactivity. This system design will support the operation of the facility during normal and abnormal conditions in providing a controlled environment designed to assure that release of radioactive particulates and/or radioactive gases to the atmosphere will be as low as reasonably achievable. Differential pressures between established zones or areas of differing radioactive contamination potential will be maintained to confine contaminants such that the flow of air will be from areas of lesser to areas of greater contamination potential.

The design of the fire protection system will provide:

1. Fire detection systems that will give early warning signals for personnel safety.

2. Systems for the prompt extinguishment of fires as they occur.
3. Necessary fire protection for filtration systems and other safety-related equipment to prevent the release of significant amounts of potentially radioactive contaminants to the atmosphere.

3.7.5 Balance of Plant Electrical Systems

The BOP electrical systems that will be required to generate main electrical power, auxiliary power, emergency power, to transform and transmit power, and provide other power plant support services are as follows:

- Main electrical generation
- High voltage switchyard
- Auxiliary power supply and distribution
- Electrical on-site emergency generation
- Direct current and uninterruptible AC power supply
- Protective relaying system and electrical control
- Lighting
- Communication
- Alarm and annunciation

The main electrical generator will produce at least 1000 MW net power output at a 0.90 power factor.

The high voltage switchyard will be designed and arranged to:

1. Transform and transmit the electrical output power from the main generator to the take-off transmission towers within the switchyard.
2. Provide power to the plant's auxiliaries, both normal and emergency, whenever the electrical generator is shut down for any reason.
3. Interconnect with the local electrical utility at the transmission take-off towers within the switchyard.

The auxiliary power supply and distribution systems will be arranged to provide all auxiliaries, both normal and emergency, with power from one of the following three sources over one of two separate electrical power distribution systems.

1. Off-site power source through the switchyard and over its separate power distribution system to the major auxiliary power busses.
2. Off-site power source through the switchyard and the main and auxiliary transformers to the major auxiliary power busses when the main generator is shut down.
3. From the main generator through the auxiliaries transformer to the major auxiliary power busses.

Each emergency auxiliary power bus will be furnished power from the on-site emergency generation system which will consist of the following:

1. On-site electrical generators, possibly diesel-driven, each with sufficient capacity to provide power to all emergency auxiliaries.
2. The auxiliary power busses will be divided into two groups so that the loss of any given bus will interrupt the power to only a portion of the emergency auxiliaries.
3. Each emergency generator will be connected directly to each major bus of the emergency system.

In addition to the emergency power plant auxiliaries as outlined above, the battery chargers for the various station batteries and any ac power requirements of the uninterruptible ac power supplies will also be connected to motor control centers within the on-site emergency generation system.

The power plant's direct current and uninterruptible ac power supplies will consist of station batteries, battery chargers, inverters and associated direct current and uninterruptible power panels. All the ac power devices such as battery chargers and constant voltage transformers will be furnished power from motor control centers within the on-site emergency generation system to insure that ac power is always available to these two systems.

The protective relaying systems will be designed to protect zones of the power system with back-up relaying systems to function only if the primary zone relay system fails to function. The main transformer differential with time overcurrent back-up is an example of a protective zone. The main generator will be protected by a protective zone but also from the local utility's power system short circuit conditions, loss of field and phase to grounding circuit conditions within the generator's grounding zone.

Volts/Hertz protection of the generator and main transformer will also be provided. The protective relaying for the outgoing power lines, from the switchyard, will be designed to meet a local utility's power system requirements. Within the power plant the various auxiliary power busses will be included in protective zones with the power system feeders protected by phase overcurrent relays with ground overcurrent back-up.

The alarm and annunciation system will include data acquisition and storage, radiation monitoring, and event recording of major problems and annunciation in various locations of the problems involved. The operation of any electrical alarm point will be annunciated locally, with a combination of these points being annunciated in the main control room under a title giving the type and location of the trouble. This will include the power plant, the switchyard and other areas within the site. The alarm and annunciation system will include plant security.

3.7.6 Trade-Off Studies and Investigations

During fiscal 1980 many trade-off studies and investigations will be undertaken with respect to the balance of plant in order to determine the best concepts. Obviously, within the funding limitations, these investigations will be more in the form of scoping studies than optimization studies. Some of these studies that will be undertaken are briefly described below.

1. Evaluate the site area requirements based on the exclusion area boundary distances from the postulated tritium and other radioactive materials release assessments.
2. An investigation will be made to determine steam conditions and final cycle arrangement for optimum thermal efficiency. This will be conducted with full cognizance of the reactor circulating coolant conditions.
3. A trade-off study will be made as to the most economical and efficient turbine-generator arrangement.
4. The condensate cycle arrangement will be investigated in conjunction with the steam cycle analysis to determine component sizes, and configuration for efficient plant arrangement and reliable operation.

5. The reactor systems and steam generator operation will be investigated to determine the extent of residual heat removal required during normal shutdown and transient by reactor trips and other abnormal operating conditions to determine what additional auxiliary cooling water requirements should be incorporated into the design to assure a maintenance free shutdown of the reactor.
6. An evaluation will be made to determine the optimum source of steam within the cycle for the various users.
7. A trade-off study will be performed to determine the desirability of recovering and storing the inventory of nitrogen from the cryogenic system during shutdown and/or abnormal conditions.
8. A trade-off study will be made as to the advantages and disadvantages of connecting the main electrical generator directly to its main transformer without a generator breaker.
9. Investigate the advantages and disadvantages from the standpoint of reliability and cost of various switchyard arrangements.
10. Compare the functions, purposes, advantages and cost for two completely independent auxiliary power distribution systems with two off-site power sources vs a single power distribution system with only one off-site power source.
11. Investigate the possibility, the advantages, disadvantages and cost of installing harmonic filters and inverter equipment in the switchyard or a central location within the plant for use with the energy-storage devices.
12. Evaluate the advantages and disadvantages of several facility layouts and arrangements. This will include evaluating means of minimizing the size of the reactor building.

References

1. B. Badger, et al., "UWMAK-I, A Wisconsin Toroidal Fusion Reactor Design," University of Wisconsin, UWFD-68 (1975).
2. B. Badger, et al., "UWMAK-II, A Conceptual Tokamak Power Reactor Design," University of Wisconsin, UWFD-112 (1975).
3. R. W. Conn, et al., "UWMAK-III, A Noncircular Tokamak Power Reactor Design," University of Wisconsin, EPRI-ER-368 (1976).
4. R. G. Mills (ed.), "A Fusion Power Plant," Princeton Plasma Physics Laboratory, MATT-1050 (August, 1974).
5. "General Facility Considerations," The Ralph M. Parsons Company, GA-A15100, Vol. VII (October, 1978).
6. "Engineering Support - Facility Studies," The Ralph M. Parsons Company, GA-A14614, Vol. X (October, 1977).
7. "Conceptual Design Study of an Experimental Power Reactor Facility for a Tokamak Type Fusion Reactor," The Ralph M. Parsons Company (1976).

3.8 Licensing and Safety

3.8.1 Licensing

Since fusion power development is still undergoing both scientific and engineering demonstration, the types of anticipated licensing considerations and the extent of imposed regulatory requirements on the tenth commercial fusion power plant are subject to varied assumptions. Distinct characteristics inherent in a fusion reactor, including a lack of fission products and actinides, as well as an absence of potential for runaway nuclear reactions, will strongly suggest that the commercial application of fusion power may avoid a lengthy and complicated licensing process.

During FY 1979, emphasis in licensing evaluation has been on the identification of licensing requirements which could reasonably be expected for the construction and operation of a commercial fusion power plant after fusion, as an energy source, has reached a mature status. Reported herein are preliminary results of some of the evaluations. Several studies, including an assessment of potential licensing schedules, are still in progress and will be part of the FY 1980 project effort.

3.8.1.1 Licensing Considerations

There are basically two fundamental approaches to the evaluation of licensing requirements for a future commercial fusion power plant. The first approach is to consider the licensing requirements from an "absolute" basis. That is, to come up with a set of licensing requirements independent of the current licensing practices and requirements. The second approach is to project the future fusion licensing requirements from "how it has come about to be where we are today." That is to project the future requirements from what we know about the development of the current licensing framework. This second approach includes deriving a set of fusion licensing considerations from the current licensing process and requirements given the necessary considerations that the fusion reaction and its inherently safe nature and that potential licensing issues are substantially different from those of nuclear fission power plants. The second approach is being pursued on STARFIRE because of the limited resources to be applied to this

task. The first approach is likely to be required in the future after well defined concepts are available.

Several licensing considerations have been identified including the following:

1. A federal regulatory agency (or group of agencies) will be empowered to protect the safety of plant operating personnel, protect the safety of the public beyond site boundaries, and insure minimum adverse environmental impacts.
2. Such a federal regulatory agency will perform an in-depth review of the owner's application for aspects of site, design, construction, and operation.
3. To reflect the inherent public safety advantages of fusion. This action, in turn, should lower the plant design and construction costs, as well as reduce the overall project schedule, including licensing.
4. The future fusion application format will be a modification of the review requirements presently mandated for fission projects.
5. Presently known problems concerning such aspects as tritium, radiation damage to materials, remote maintenance, lithium reactivity, biomagnetics effect, solid radwaste disposal, and other problems yet to be detected will have been either resolved or controlled in some realistic manner. Such an assumption is fundamental to a mature fusion industry.

3.8.1.2 Licensing Schedule

It is a basic premise of licensing for future fusion projects that some federal agency will act as a central point of coordination for receipt and review of all fusion license applications. "One-step licensing" is not believed to be a realistic approach to such licensing, and therefore it is anticipated that a two-step licensing activity for construction and operation, respectively, will be required.

Since safety analysis reports and environmental reports have been well established as review tools, similar documents in some form similar to the present Preliminary Safety Analysis Report (PSAR) for construction, the Final Safety Analysis Report (FSAR) for operation, and the Environmental Report (ER) for both construction and operation can be expected. Review of these documents is likely to require a minimum of 18 months during the construction permit process for legal, technical, environmental, and public comments. The subsequent operating license process could require approximately the same length of time.

3.8.1.3 Applications Format

The existing NRC Regulation Guide 1.70 is the type of structured information which will be required for fusion licensing and safety review. Certain definitive modifications, however, can be expected as a minimum and will include the following:

1. Deletion of fission-related aspects such as containments, core cooling, fission product removal, etc.
2. Extensive additions to in-plant electrical system details applicable to fusion.
3. Revisions to waste management concepts to reflect fusion radwaste requirements.
4. Expansion of in-plant protection systems to identify fusion-related aspects such as high magnetic flux and cryogenic systems.
5. Complete redirection of accident analysis based upon fusion conditions.

Within a mature fusion industry, it can be expected that the application data will reference previous fusion Safety Analysis Reports (SAR's) and the depth of detail will thereby increase with each succeeding SAR. Also, it is to be expected that the coordinating government agency will be structured along the lines of the SAR format to facilitate parallel review of application material.

3.8.2 Safety

In conjunction with establishing the design parameters for the commercial tokamak, first-order safety assessments are being performed. This is being done in order that potential safety hazards and environmental impacts may be identified and either be avoided or minimized to the extent practicable. Hazards to those persons who will be working in the fusion power plant, as well as to the general public, are being considered.

The utilization of radioactive tritium, which has a half life of 12.3 years and emits a weak beta particle ($E_{\text{max}} = 18.6 \text{ keV}$), is one of the safety hazards under consideration. The design of the tritium recovery and processing systems for both the plasma and the breeder blanket will be developed in order to minimize the total amount of tritium involved, especially the vulnerable portion that would be in a form or a location where it would be readily released in an accident. The amount of tritium to be kept in storage will also be minimized.

A total of 2500 g of tritium will be the inventory on the site. Of that total, 755 g in the reactor building and 230 g in the tritium facility building are considered to be vulnerable -- that is, could conceivably be released in the case of an accident. The invulnerable portions are some 1000 g tied up in the solid breeder material and 500 g tied up in the storage on gettered beds.

The induced radioactivities in the first wall and the structural materials in the breeder blanket and shield are the other major sources of radioactivity presently being considered. Several candidate materials have been considered. The primary candidates are 316 stainless steel, ferritic stainless steel (CE), Ti 4381 and V-15Cr-5Ti alloys. The normalized activities of these alloys have been calculated for the STARFIRE design conditions both during reactor operation and following reactor shutdown. See Sec. 3.2.3 for the details of those calculations. The total decay heat, the beta decay heat, and total biological hazard potential in air have also been calculated for a number of time steps over the same range of values as for the normalized activities.

The atmospheric dispersion of tritium and of structural material was examined for a hypothetical accident based on a short-term uniform release from a point source. A simple model is utilized in order to compare the relative external dose from the candidate alloys dispersed in a Gaussian plume. The impact of tritium leaking from the reactor building following a hypothetical accident was also calculated using the same methodology.

The relative external doses due to structural material being volatilized by a hypothetical reactor accident and leaking out of containment are shown in Fig. 3.8-1. Based on these results, 316 SS, Ti 4381 and ferritic SS (CE) yield approximately equal doses, while the V-15Cr-5Ti alloy is about one-fifth as great as the others. These doses are based upon extremely conservative assumptions and are for ground-level exposures at the centerline of the plume.

The relative effect of the vulnerable tritium inventory in the reactor building was also calculated. Its normalized activity would be 2.4×10^3 Ci/MWth and the external whole body dose from the plume at 200 meters and at one hour following an accident would be 5.7×10^{-10} rem/s MWth. Thus the effect of tritium on dose compared to that due to structural material is small -- approximately 1.0% that of V-15Cr-5Ti. The biological hazard potential of the accidentally released tritium in air would be 12 km^3 air/MWth and in water would be $8 \times 10^{-4} \text{ km}^3$ water/MWth. The BHP of tritium in air is much less than that of the structural material. It is approximately 0.05% that of V-15Cr-5Ti which is the candidate with the lowest BHP air.

To date no effort has been made to postulate details regarding mechanisms for making the radioactive materials within the reactor mobile or volatilizing them. Nor have mechanisms for breaching multiple layers of containment in order to release the materials into the environment been established. At a later time various initiating events may be postulated, event trees developed and analyzed. The mechanical design of the first wall, blanket, and shield needs to be better established before such studies can be meaningfully done.

The extent of overpressurization of the reactor building due to the accidental release of helium from either the inventory associated with the superconducting magnets or a design utilizing helium as the primary coolant, and due to the release of vulnerable tritium have all been calculated to a first approximation.

Figure 3.8-1. Relative external dose from plume vs distance (STARFIRE Design).

The helium inventory in the superconducting magnets is approximately 50 tonnes and the containment building volume is about $2.0 \times 10^5 \text{ m}^3$. Thus the pressure in the reactor building would be 0.2 atm following an accident in which 10% of the helium is lost and assuming the peak containment temperature is maintained at 300 K.

For the STARFIRE design, in which the first wall is cooled with water and the blanket is cooled with helium, the overpressurization in the containment building following a LOCA in both coolant systems would be about 0.2 atm (3 psi).

If water were used as the coolant in both the first wall and blanket, and a pipe break were to occur, the overpressurization of the containment would be about 1.25 atm based upon similar computations for light water fission reactors.

Likewise, because of the intentionally small mass of vulnerable tritium in the reactor building (755 g), an accidental release of all of it would result in a negligible overpressurization of 3×10^{-5} atm.

Incorporation of an Emergency Air Detritiation System⁽¹⁾ in the plant design would provide an engineered safety system for reducing the amount of tritium which would leak out of the reactor building or out of the tritium facility after an accidental release of tritium in either building.

Reference

1. R. Clemmer, "TCODE -- A Computer Code for Analysis of Tritium and Vacuum Systems for Tokamak Fusion Reactors," Argonne National Laboratory, ANL/FPP/TM-110 (August, 1978).

APPENDIX

Summary of STARFIRE Workshop

1. Session on Plasma Engineering

Participants

Chairman, Dan Cohn, MIT

R. Conn, U. of Wis.
G. Emmert, U. of Wis.
R. Jacobsen, PPPL
L. Lao, U. of Wis.
D. McAlees, Exxon
M. Peng, ORNL
P. Rutherford, PPPL
C. Singer, PPPL
W. Stacey, Georgia Tech.
T. Yang, MIT
S. Yoshikawa, PPPL

RF Current Drive

It is believed that the burn phase of rf steady-state current drive is an acceptable and appropriate choice for the STARFIRE design. Among the general concerns are that the high Q operation is very sensitive to the physics assumptions; the experimental data base is meager, and the hollow current profiles have considerably less credibility than centrally peaked profiles. With regard to rf current drive during the starting phase, this is regarded as somewhat less credible than the burn phase, and also not as necessary. The following general comments and recommendations have been made.

- Since accessibility improves at a higher field, compute the maximum field versus rf power to get centrally-peaked current profiles.
- Consider the start-up phase with ECRH and/or supplemental OH drive to initiate the plasma and possibly to induce $\sim 1\text{--}2$ MA of current prior to the rf drive. Estimate the phase shift capability of the LH system to find the initial temperature for LH operation; compute the available volt seconds in the EFC system, and determine what ohmic coils, if any, are needed to assist startup.

In view of the speculative nature of rf current drive with lower hybrid heating, a high priority should be given to examining alternative current drivers:

- Magnetosonic waves represent the best combination of low power requirements and broad current profiles. Relativistic electron beams are promising, but the ideas are not as well developed.
- A useful calculation would be the cost of energy for different driver options. Unfortunately, this is probably impossible considering the differing levels of understanding of the constraints and reliability for various options.

RF Heating

The concerns with rf heating, per se, as opposed to current drive include the question of preferential edge heating and overall thermal stability control. It is recommended that the thermal stability of the high Q rf driven system with enhanced radiation be evaluated as planned.

Impurity Control

It was agreed that it was appropriate for STARFIRE to incorporate a nondivertor approach to impurity control and that most of the effort should be spent on this approach. As such, the limiter system should receive a major design effort. There were no objections raised to the physics basis of the limiter design. A point that may be of substantial help to the limiter design is that the energy fall-off rate may be faster than the particle fall-off rate in the scrape-off zone, thus permitting a reduction in peak heat load by appropriate shaping of the limiter surface. There is concern with the erosion of the low-Z limiter coating by sputtering, particularly in the sensitivity of sputtering calculations to the rate and uniformity of redeposition. This should be looked at in more depth, if possible.

Divertors should be considered as a backup option, but only those that can be accommodated without major change in the STARFIRE engineering design. There is concern that a bundle divertor may be incompatible with rf current drive because of the expulsion by ripple effects of current carrying electrons. This effect should be assessed, if possible.

MHD Equilibrium

The main concern was that the equilibrium profiles be consistent with that expected from the rf current drive. It was decided that a self-consistent calculation of transport and of heat and momentum deposition be integrated with the stability and equilibrium calculations. The ability to do plasma shaping in order to obtain satisfactory critical betas was emphasized and refinements of the EF coil system to accomplish this, as well as startup, seem to be needed. In general, the magnitude of beta as discussed for STARFIRE appear reasonable.

Hollow plasma current density profiles may not be entirely credible and work needs to be done on their evolution and stability. Peaked profiles at higher magnetic fields may be more credible.

Control of plasma position on a short time scale is necessary, but the time constant is not certain. It may be possible to transfer the bulk of the magnetic energy released during a disruption into a system of shorting rings placed around the plasma. Disruptions will probably be slowed down by the proximity of the first wall/blanket, but control of major disruptions may be impossible. We should assume that plasma diagnostics will be sufficiently advanced by the time STARFIRE is built and that the f-profile can be continuously monitored and that the rf can be used to modify the current profile in order to avoid disruptions. Major disruptions may be considered as an accident mode which occurs mainly due to control system failure. Clearly, the tokamak must be constructed so as to accept at least a small number of major disruptions.

2. Session on Blanket Engineering

Participants

Chairman, J. Scott, ORNL

J. Anderson, LASL
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R. Gold, Westinghouse
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T. Huxford, ORNL
M. Kazimi, MIT
G. Kulcinski, U. of Wis.
L. Muhlestein, HEDL
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P. Sager, GA
D. K. Sze, U. of Wis.
W. Wiffen, ORNL
B. Woelfer, U. of Wis.

The blanket engineering workshop was divided into three subgroups, which addressed the following topics:

- Breeding Materials
- Structural Materials
- Mechanical Design

Key issues considered and opinions of the workshop participants are given for each topic. Two general recommendations from the blanket engineering session are also given.

Breeding Materials

1. Issue - Is Li_2O more promising than other solids?

Opinion - Yes. No need for multiplier (?). High melting point and wider operating temperature window were cited.

2. Issue - Is data base adequate?

Opinion - No! Key deficiencies need to be identified by STARFIRE team and programs initiated to fill needs, e.g., thermophysical properties, irradiation response.

3. Issue - Will Li_2O perform as perceived in STARFIRE conceptual design?

Opinion - Radiation sintering effects may not be as limiting as presently perceived by STARFIRE team.

- Mass transfer limitations look critical. Present tritium removal strategy probably would not work as is. Tritium inventory may be $\gg 1$ kg at steady state.

4. Issue - Can Li_2O breeder material be produced in large quantity for present STARFIRE design concepts?

Opinion - With present-day methodology, probably not. Preparative capabilities need to be developed and requirements resolved.

5. Issue - Is structural separation of Li_2O from primary coolant helium necessary?

Opinion - Yes. Impurity carryover to breeder, corrosion and fouling problems cited.

6. Issue - Is Li_2O /structure compatibility a problem?

Opinion - Yes. We need data on compatibility of Li_2O (dry and wet) with candidate alloys.

7. Issue - Are the perceived safety advantages of ceramic breeders over liquid lithium and alloys realistic?

Opinion - Yes, if the tritium inventory can be controlled.

8. Issue - Are the proposed breeder subassembly designs fabricable?

Opinion - Yes. The breeder-in-tube design is probably better from the standpoint of attaining high-breeder material density and loading of the breeder module.

Structural Materials

1. Issue - Should activation be a design/selection guideline?

Opinion - Issue is related to storage and reprocessing.

- Within ~ 50 -year time period, there appears to be very little choice between candidate ferritic or austenitic structural materials.

- Not a realistic selection issue.
 - Group feels that alloy tailoring to reduce activity is credible.
 - Group feels isotopic tailoring is not credible.
2. Issue - Is 28 MW-y/m² a reasonable lifetime goal?
- Opinion - Probably unrealistic based on current data base.
- Relaxation of goal may well permit selection of a different structural material (e.g. change to austenitic stainless steel).
 - Group did not make specific recommendation for change.
3. Issue - Was materials selection appropriate for STARFIRE?
- Opinion - Agree candidate list was comprehensive.
- Initial groundrules - namely, a prior selection of coolant and breeder materials - restricts structural materials options.
 - Agree prime candidates are ferritic and austenitic steels (for first wall and probably blanket); Ti alloys may be acceptable for limited blanket designs.
 - Recommend/endorse selection of ferritics for both blanket and first-wall applications; a major driver for this selection is need for assessment of reactor design/analysis/performance which results from use of ferritic steels.
 - Consider that choice of ferritics over austenitics reflects a vote for evaluating potential long-term, long-life pay-off versus designing with the firmer data base of austenitics.
4. Issue - Materials design recommendations.
- Opinion - Select a specific reference alloy (e.g. HT-9)
- Factor into design the probable or possible effects of irradiation, e.g.
 - Assume a DBTT > ? ft.-lbs.
 - Assume a lower shelf energy of < ? %
 - Assume some swelling - ? %

- Incorporate into first wall and blanket design features for:
 - Post-weld heat treatment schedules
 - Use of ferritic/nonferritic transition joints
- Assess implications of:
 - Ferromagnetic behavior for both normal and off-normal conditions
 - Possible radioactive mass transfer and/or tube plugging due to corrosion

Mechanical Design

1. Issue - Is coolant selection appropriate?
 Opinion - Yes. Coolant selections are endorsed for the reference, alternate and back-up for the present FW/B configuration.
2. Issue - Is 41% power cycle efficiency correct?
 Opinion - Validity of 41% efficiency value is questionable in light of prior experience. The efficiency value should be examined in greater depth.
3. Issue - What are impacts of D₂O cost and requirements?
 Opinion - Requirements for and cost of D₂O should be examined to determine significance. CANDU information should be considered.
4. Issue - Superiority of FW/B integration versus separation.
 Opinion - Possibility of integrating first wall and blanket into a single unit should be reexamined. ORNL design achieved 4.0 MW/m² with helium coolant. Integration should lead to simpler design overall.
5. Issue - Coolant flow path for pressurized tube concept.
 Opinion - Configuration and flow directions for the pressurized concepts are endorsed. Radial flow direction would lead to much greater number of tubes and more difficult thermal-hydraulic problems.

6. Issue - Combination of blanket design alternatives.
- Opinion - Two concept options were examined and the reviewers support the project's program to pick one of these. Careful attention should be given to thermal stresses, fatigue considerations and constraints.
7. Issue - Structural integrity of pressurized module concept.
- Opinion - Additional analysis should be performed to confirm structural integrity, particularly in design of pressurized module shell.
8. Issue - Ferromagnetic and seismic loads should be examined for their effects on the first wall, blanket and shield and on the limiter in particular.
9. Issue - Number and size of blanket modules.
- Opinion - Methods should be explored to reduce the number of different modules per sector. Fabricability, maintainability and spares inventory are affected.
10. Issue - Fault tree analysis has not been performed.
- Opinion - The blanket should examine probable fault conditions for limiter, first wall, blanket and shield. Consider, also, how they could be detected, both before and immediately after occurrence.
11. Issue - Magnitude of limiter heat load.
- Opinion - Development of limiter should be identified as critical R&D need.
12. Issue - Consideration of off-normal conditions.
- Opinion - Off-normal conditions should be analyzed for selected design options.

General Considerations

1. The STARFIRE team should set as a goal a workable reactor using a solid breeder and should identify the R&D needed to resolve associated major issues.
2. The community in general should take care in making statements about lithium systems, so that liquid lithium systems are not tagged as "unsafe".

3. Session on Design Integration, Magnets and Maintenance Review

Participants

Chairman, T. Shannon, ORNL

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L. Masson, EG&G
P. Reardon, PPPL
M. Sniderman, Westinghouse
P. Spampinato, Grumman
I. Sviatoslavsky, U. of Wis.
K. Wakefield, PPPL
N. Young, Ebasco

The design integration, magnets and maintenance review committee was chaired by Tom Shannon of ORNL. The team members consisted of senior engineers from industry, laboratories and universities experienced in tokamak design, development and construction. The review committee was favorably impressed with the accomplishments of the program to date, and they complimented the design team for the manner in which the review was organized. The information was presented exceptionally well both in the documents and in the oral presentations.

The evaluation approach used was to focus the discussions into four categories: 1) design approach, 2) mechanical/structural design, 3) vacuum/tritium/thermal hydraulics, and 4) maintenance/assembly.

A brief discussion of the findings is as follows:

Design Approach - The committee believes the program objectives would be best served by expanding the systems engineering aspects of the program. Recommendations include that the program goals for FY-1980 be reviewed and a clear set of requirements and objectives be established. Tasks should be structured according to the necessary priorities and within resource limitations.

Major areas the committee recommends STARFIRE focus its activities on are: reliability, availability, maintainability and the innovative engineering areas, which include the divertor/limiter and design for remote maintenance.

The committee feels the reactor engineering issues should be stressed over areas such as balance of plant and tritium processing. The detail

level of TF coil design appears to be adequate for the program and little further work is recommended.

Mechanical/Structural Design - The overall structural design requires further development to establish credibility. The out-of-plane magnet loads are of particular concern.

The TF coil magnet design is of concern because of the complexity of providing steps in the helium dewar and curvature in the coil structure. Cooldown costs may also be significant and warrant a further look.

The limiter should receive a significant design effort that accounts for both normal and abnormal load conditions. Lifetime assessment is required in conjunction with an investigation of alternate maintenance methods.

Vacuum/Tritium/Thermal Hydraulics - The committee recommends that more analysis be performed in this area. Major concerns include: tritium containment under accident conditions, cleaning of the vacuum boundary, maintenance of high vacuum requirement in the plasma chamber, leakage by mechanical disconnects inside the vacuum boundary, and neutron streaming through vacuum pumping passageways.

Maintenance and Assembly - Maintenance and assembly require more attention in the program. It is believed to be the most important aspect of STARFIRE. Availability of the plant must be defensible.

The modular, all-remote approach sounds good, but requires the necessary layouts and analysis to demonstrate its practicality. The number of TF coils should be reassessed to determine if the maintenance features can be enhanced.